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Volume II--Working Papers: Nuclear Power in an Age of Uncertainty

Part 2: Issues Affecting the Viability and Acceptability of Nuclear Power Usage in the United States; The Future of Conventional Nuclear Power: Nuclear Reactor Reactor Reactor Regulation; Nuclear Reactor Regulation and Nuclear Reactor Safety; Case Studies on Regulatory Requirements.

November 1984

This report is an OTA Working Paper, which was used in the preparation of OTA's final assessment report. OTA makes these working papers available for the use of readers desiring a more detailed or technical discussion of an issue than can normally be accomodated in our final report. As an OTA working paper, this report has not been reviewed or approved by the Technology Assessment Board. The findings and conclusions expressed in the report are those of the authors and do not necessarily reflect the views of OTA, the advisory panel or the Technology Assessment Board.

J952-61

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ISSUES AFFECTING THE VIABILITY AND ACCEPTABILITY OF NUCLEAR POWER USAGE IN THE UNITED STATES

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FOR USE IN CONJUNCTION WITH:

WORKSHOP ON TECHNOLOGICAL AND REGULATORY CHANGES IN NUCLEAR POWER

OTA CONFERENCE CENTER
600 PENNSYLVANIA AVENUE, SOUTHEAST

DECEMBER 8 & 9, 1982

REPORT SUBMITTED: DRAFT - DECEMBER 2, 1982
FINAL - DECEMBER 28, 1982

ISSUES AFFECTING THE VIABILITY AND ACCEPTABILITY OF NUCLEAR POWER USAGE IN THE UNITED STATES

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SECTION 1

INTRODUCTION - THE FUTURE OF NUCLEAR POWER IN THE UNITED STATES

1.1: PURPOSE

This report presents the authors' identification and assessment of certain issues potentially affecting the future use of the nuclear power option in the United States. This report was prepared by MHB Technical Associates (MHB) of San Jose, California as authorized by the U.S. Congress Office of Technology Assessment (OTA) under contract 333-2690.0.

This report is not offered as an exhaustive study of the possible impediments to the use of nuclear power but is rather intended as a background paper for use in the December 8 & 9 workshop being conducted at the OTA Conference Center. The authors' recognize that the basis for all of the conclusions stated may not be fully documented in this report. The broad range of topics discussed in the short time available did not permit the extensive bibliographical description that might be desired. The significant issues identified and conclusions stated herein are stimulated by several decades worth of experience in the commercial nuclear power field that have been further distilled by participation in numerous Federal and State proceedings.

1.2: REPORT FORMAT

This report focuses on five issues potentially impacting the viability of nuclear power in the future. These fives issues and the report sections in which they are discussed are:

- Present Reactor Safety/Reliability Concerns - Section 2
- Industry Approach to Reactor Management - Section 3
- Nuclear Alternatives - Section 4
- Reactor Size - Section 5
- Licensing Reform - Section 6

Summaries and conclusions for each of these issues are contained at the end of each section. A general summary of the total report follows in Section 1.3. A copy of the OTA Contract Work Statement is included as Appendix A.

1.3: SUMMARY

Conclusions stated with regard to the five major topics covered in this report are as follows:

- Present generation reactor safety/reliability concerns

  1 - Present generation reactors may be adequately safe but that safety has not been adequately proven and demonstrated. If it

is adequate, it is the result of a good bit of luck rather than the result of a systematic and disciplined process.

2 - The reliability of the present reactor generation is far below desired expectations.

3 - Safety and reliability concerns in need of resolution are:

- Resolution of unresolved safety issues.
- Solution to steam generator problems.
- Corrective action for primary system failure modes (PTS, pipe cracks, & internal component cracking).
- Finalization of ATWS fix.
- Improved PWR water level measurement and assurance of inadequate core cooling indication.
- Reduction of radiation level buildup.
- Decisions regarding core damage mitigation.
- Improvements in human factors.
- Further definition of automatic safety system requirements.
- Improved reactor siting criteria.
- Improved separation and independence of safety and control functions.

- Improved criteria for common-cause and single-failure design.
- Adequate assessment and response to aging and obsolescence problems.

The following Table 1-1 presents our assessment of the relative seriousness of these concerns, the difficulty of resolving them, and the degree to which critics might agree that the problem has been, or is close to being resolved.

- Assessment of industry approach to reactor plant management

  1 - The industry approach to design, construction, operation and regulation of nuclear plants has been inadequate. This is evidenced by the recent Quality Assurance "breakdowns" at various plants.

  2 - An independent assessment of design and construction practices is recommended as a required safety assurance step prior to Operating License.

  3 - An effective independent assessment of the operational QA plan should be performed on a periodic basis.

TABLE 1-1

EVALUATION OF SAFETY/RELIABILITY CONCERNS

CONCERN	RELATIVE SERIOUSNESS H=High M=Medium L=Low	DIFFICULTY OF RESOLUTION H=High M=Medium L=Low	CRITICS AGREEMENT ON RESOLUTION P=Probable I=Improbable
System/Hardware			
USI's	H	H	I/P
Steam Generators	H	M	P
Primary System Failures	H	H	I
ATWS	M	L	P
Water Level & ICC	H	M	P
Radiation Level	M	H	I
Core Damage Mitigation	M	H	I
Human Factors	M	M	P
Criteria/Practice			
Automation Criteria	M	M	P
Siting	L	L	P
Separation & Independence	M	M	P
Common Cause/Single Failure	M	H	P
Aging/Obsolescence	H	H	I

- Evaluation of Nuclear Alternatives

  1 - All alternatives should be evaluated, not just different nuclear cycles.

  2 - None of the alternate reactors would defuse the broad concerns of radioactive waste disposal and assured control of potential weapons material.

  3 - The PIUS concept appears to have the most promise for attaining public acceptance, having been designed from the base of passive safety.

  4 - The PIUS reactor would take substantial time and financial investment to bring it to a viable product.

- Reactor Size

  1 - There appears to be no technical safety advantage to a small plant but emergency planning for smaller plant accidents might be easier to accomplish.

  2 - The overall (lifetime) cost of a range of plant sizes does not appear to have been adequately studied. We conclude the optimum size is probably smaller than sizes currently being built.

3 - There are a number of institutional and philosophical factors that tend to argue for smaller nuclear plant sizes.

- Licensing Process

1 - The existing licensing process has been beneficial and has apparently served (so far) to protect the public.

2 - There is a benefit to public participation in the hearing process.

3 - A combined CP and OL process would be beneficial if properly conducted.

4 - Alternates to the formal process should be pursued. These would include informal public meetings, required independent assessments, and ongoing participation of local government in plant reviews.

SECTION 2

PRESENT GENERATION REACTOR SAFETY AND RELIABILITY CONCERNS

2.1: INTRODUCTION

The present reactor generation has been the subject of much discussion with regard to the adequacy of its design and hardware. A substantial amount of this reassessment was initiated by the Three Mile Island Accident and subsequent reviews by various industry and regulatory agency committees. The key question however, is whether or not these reviews have addressed the correct (highest priority) issues and if the reviews have actually been turned into "fixes" at existing reactors. That question unfortunately remains to be answered by time.

Our assessment of the present reactor generation and identification of remaining concerns follows. It is important to emphasize that concerns discussed in this Section are primarily hardware or design criteria oriented. Another overlay of concern is found in the following sections of this report. In particular Section 3, which deals with the design-construction-operation-regulation process, discusses factors which play a key role in the corrective actions needed to address most of these concerns.

2.2: BASIS OF REACTOR SAFETY CONCERNS

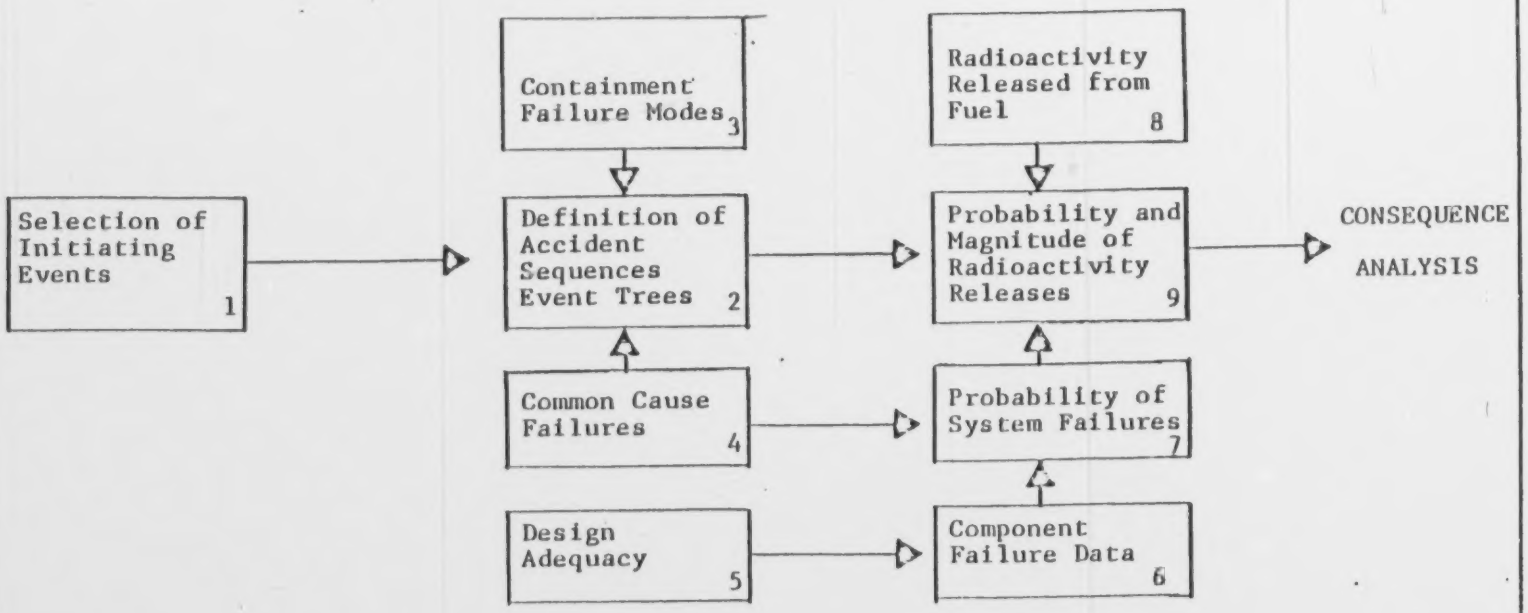
LWR safety for the most part must be judged by the potential for accidents leading to the uncontrolled release of radioactive materials. Countermeasures to prevent or mitigate such releases can include reduction of event probabilities or the mitigation of consequences. However, if public acceptance of nuclear power is to be achieved, event occurrence must be minimized. This does not mean that effective emergency plans need not be available, they must. Public perception however, will be most affected by whether or not a potentially serious accident is permitted to develop. This means then that accident probability must be given a priority ranking in this review.

The most extensive probabilistic analysis of reactor accidents is in the Reactor Safety Study, WASH-1400. The WASH-1400 model is of interest because it identifies areas where safety concerns may arise. Figure 2-1 shows a breakdown of the model used in quantifying probability of reactor accidents leading to radioactive releases. The first four steps of the model involve the definition of accident initiating events and accident sequences including failure modes in the containment and common-cause failures. Steps 5-7 of the process

FIGURE 2-1

SUBTASKS IN QUANTIFYING THE

PROBABILITY OF RADIOACTIVE RELEASES


Containment Failure Modes 3
Radioactivity Released from Fuel 8
Selection of Initiating Events 1
Definition of Accident Sequences Event Trees 2
Probability and Magnitude of Radioactivity Releases 9
CONSEQUENCE ANALYSIS
Common Cause Failures 4
Probability of System Failures 7
Design Adequacy 5
Component Failure Data 6

establish numerical values of the different component and system failure rates based on actual or comparable data. These are then combined in steps 8 and 9 to generate the theoretical probability and magnitude of an accidental release.

This probability model then gives a basis for determining which of many potential concerns are of most safety significance. The generic and TMI safety issues discussed in the following subsections have been screened by a number of groups using this basis to arrive at those given highest priority. Our priority assessment follows in Section 2-6.

2.3: GENERIC SAFETY ISSUES

Evidence of possible inadequacies of operating light-water reactors has accumulated in the form of "unresolved issues." For example, 133 issues were designated in 1977 by the NRC as requiring Staff attention. [1/] In January, 1979, the NRC further highlighted 17 of the items as having the highest safety impact. These issues, each a potential deficiency affecting many commercial nuclear plants, have previously been identified in a variety of documents including the NRC's Technical Safety Activities Report (TSAR), [2/] the Advisory Committee on Reactor Safeguards (ACRS) list of generic issues, [3/] and finally, the list of issues raised by concerned NRC Staff (Staff) members in response to the new NRC procedures for dealing with Staff dissent. [4/] [5/]

A large number of design and operational problems, which have been observed at commercial nuclear power plants, have been quite thoroughly investigated and appear to have been resolved. It is quite likely that many of these potential deficiencies will not impact on the operation or reliability of future plants. However, as discussed in the following portions of this Subsection, a large number of generic technical issues have also been identified, but have yet to be resolved. Identified are technical deficiencies that appear to be significant to reactor safety.

Background

The term, "unresolved safety issues" identifies deficiencies which may contribute to amplifying the risk to public health and safety from power reactor accidents. The history of the U.S. nuclear power program has shown that in many cases, the accidents which occur are not the ones which have been analyzed in the licensing process. However, after the fact, the accident initiators or major contributors, are often found to have been on someone's list of safety concerns.

The NRC's Advisory Committee on Reactor Safeguards has been publishing since December 18, 1972, a series of lists of generic items (problems) which are of concern to light-water reactors. A 1977 list includes a total of over 75 items of which around 25 are considered unresolved. 6/ Issues listed include such important and fundamental safety aspects as containment, pressure vessel, ECCS components, piping and electrical supplies. Ten of the problems have been on the list since its inception in 1972, yet remained unresolved. Even when items are declared resolved, there is no certainty that a solution has been implemented. Anticipated Transients Without Scram (ATWS) represents an example of this category in that it has been declared resolved in an administrative sense. Nearly 50 "resolved" items exist; it is not known

what percentage of them resulted in real fixes in hardware or structures and, conversely, how many were "administrative" only.

The ACRS list is by no means a complete accounting of "unresolved safety issues." The NRC in 1974 tabulated the unresolved safety issues into an internal Technical Safety Activities Report (TSAR). When released in 1975, the TSAR listed 223 items of concern to the NRC of which 173 were categorized as having "an important impact on the licensing review process." 7/ The NRC apparently discontinued the TSAR, but in the fall of 1976, in response to a memorandum by Rusche of the NRC, one or more members of the NRC staff identified 27 technical issues as "problems whose priority, progress, or resolution was, in their opinion unsatisfactory." 8/

In October, 1976, the NRC Commissioners directed the NRC Staff to develop a program plan for the timely resolution of the generic technical issues. Implementation of the NRC program began in April, 1977. Initially, each of the four NRC divisions reporting to the Office of Nuclear Reactor Regulation described and proposed those generic items it considered to warrant the highest priority. Proposals were received for 355 tasks 9/ of which, following consolidation and elimination, 133 tasks were eventually selected for review. A set of uniform criteria was applied to the generic technical

activities to establish their priority for resolution. The 41 tasks shown in Table 2-1 were classified as Category A, warranting priority attention. The remaining issues were divided into 73 Category B tasks, 17 Category C tasks, and 3 Category D tasks as shown in Table 2-2.

An additional source of information evaluating these uncertainties is documented in a preliminary risk-based evaluation of the generic issues performed in 1978 by the NRC. 10/ The primary objective of the NRC review was to identify those generic items having the greater safety significance on a relative risk basis. It was not the intent to make an absolute risk determination on the items, nor was there any intent to make judgements as to acceptable levels of risk. Using relative risk-based arguments, the 133 generic task action items were separated into four general categories: (I) those having potentially high risk significance; (II) those with potentially low risk significance; (III) those items which appear to have negligible risk potential; and (IV) those items not directly relevant to risk. Table 2-3 presents a risk-based ordering of the various issues for categories (I) and (II). 11/

TABLE 2-1

CATEGORY A GENERIC ACTIVITIES

(Original Listing in NUREG-0410)

TASK NO:	TITLE:	RELEVANT TO BWR	RELEVANT TO PWR
A-1	WATER HAMMER	X	X
A-2	ASYMMETRIC BLOWDOWN LOADS ON THE REACTOR VESSEL	X	X
A-3	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY		X
A-4	COMBUSTION ENGINEERING STEAM GENERATOR TUBE INTEGRITY		X
A-5	BABCOCK & WILCOX STEAM GENERATOR TUBE INTEGRITY	X	
A-6	MARK I SHORT-TERM PROGRAM	X	
A-7	MARK I LONG-TERM PROGRAM	X	
A-8	MARK II PROGRAM	X	X
A-9	ATWS	X	X
A-10	BWR NOZZLE CRACKING	X	
A-11	REACTOR VESSEL MATERIALS TOUGHNESS	X	X
A-12	FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS	X	X
A-13	SNUBBERS	X	X
A-14	FLAW DETECTION	X	X
A-15	DECONTAMINATION	X	X
A-16	STEAM EFFECTS ON BWR CORE SPRAY DISTRIBUTION	X	
A-17	SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS	X	X
A-18	PIPE RUPTURE DESIGN CRITERIA	X	X
A-19	DIGITAL COMPUTER PROTECTION SYSTEMS	(PLANTS WITH DIGITAL COMPUTERS)	
A-20	IMPACTS OF COAL FUEL CYCLE	(ENVIRONMENTAL)	
A-21	MAIN STEAM LINE BREAK INSIDE CONTAINMENT		X
A-22	PWR MAIN STEAM LINE BREAK - CORE AND PRIMARY COOLANT BOUNDARY RESPONSE (MSLB OUTSIDE CONTAINMENT)		X
A-23	CONTAINMENT LEAK TESTING	X	X
A-24	QUALIFICATION OF CLASS IE SAFETY-RELATED EQUIPMENT	X	X
A-25	NONSAFETY LOADS ON CLASS IE POWER SOURCES	X	X
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION (OVERPRESSURE)		X
A-27	RELOAD APPLICATION GUIDE	X	X
A-28	INCREASE IN SPENT FUEL STORAGE CAPACITY	X	X
A-29	DESIGN FEATURES TO CONTROL SABOTAGE	X	X
A-30	ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES	X	X
A-31	RHR SHUTDOWN REQUIREMENTS	X	X
A-32	EVALUATION OF OVERALL EFFECTS OF MISSILES	X	X
A-33	NEPA REVIEWS OF ACCIDENT RISKS	(ENVIRONMENTAL)	
A-34	INSTRUMENTS FOR MONITORING RADIATION AND PROCESS VARIABLES DURING ACCIDENTS	X	X
A-35	ADEQUACY OF OFF-SITE POWER SYSTEMS	X	X
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	X	X
A-37	TURBINE MISSILES	X	X
A-38	TORNADO MISSILES	X	X
A-39	DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC	X	
A-40	SEISMIC DESIGN CRITERIA - SHORT-TERM PROGRAM	X	X
A-41	SEISMIC DESIGN CRITERIA - LONG-TERM PROGRAM	X	X

TASKS A-42, A-43, & A-44 WERE ADDED TO THE LIST IN NUREG-0510.

TABLE 2-2

CATEGORY B TECHNICAL ACTIVITIES

TASK NO:	TITLE:	RELEVANT TO BWR	RELEVANT TO PWR
B-1	ENVIRONMENTAL TECHNICAL SPECIFICATIONS	X	X
B-2	FORECASTING ELECTRICITY DEMAND	X	X
B-3	EVENT CATEGORIZATION	X	X
B-4	ECCS RELIABILITY	X	X
B-5	DUCTILITY OF TWO-WAY SLABS AND SHELLS AND BUCKLING BEHAVIOR OF STEEL CONTAINMENTS	X	X
B-6	LOADS, LOAD COMBINATIONS, STRESS LIMITS	X	X
B-7	SECONDARY ACCIDENT CONSEQUENCE MODELING		X
B-8	LOCKING OUT OF ECCS POWER OPERATED VALVES	X	X
B-9	ELECTRICAL CABLE PENETRATIONS OF CONTAINMENT	X	X
B-10	BEHAVIOR OF BWR MARK III CONTAINMENT	X	
B-11	SUBCOMPARTMENT STANDARD PROBLEMS	X	X
B-12	CONTAINMENT COOLING REQUIREMENTS (NON-LOCA)	X	X
B-13	MARVIKEN TEST DATA EVALUATION	X	X
B-14	STUDY OF HYDROGEN MIXING CAPABILITY IN CONTAINMENT POST-LOCA	X	X
B-15	CONTEMPT COMPUTER CODE MAINTENANCE	X	X
B-16	PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT	X	X
B-17	CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS	X	X
B-18	VORTEX SUPPRESSION REQUIREMENTS FOR CONTAINMENT SUMPS		X
B-19	TERMAL-HYDRAULIC STABILITY	X	X
B-20	STANDARD PROBLEM ANALYSIS	X	X
B-21	CORE PHYSICS	X	X
B-22	LWR FUEL	X	X
B-23	LMFBR FUEL	LMFBR's	
B-24	SEISMIC QUALIFICATION OF ELECTRICAL AND MECHANICAL EQUIPMENT	X	X
B-25	PIPING BENCHMARK PROBLEMS	X	X
B-26	STRUCTURAL INTEGRITY OF CONTAINMENT PENETRATIONS	X	X
B-27	IMPLEMENTATION AND USE OF SUBSECTION NF	X	X
B-28	RADIONUCLIDE/SEDIMENT TRANSPORT PROGRAM	X	X
B-29	EFFECTIVENESS OF ULTIMATE HEAT SINKS	X	X
B-30	DESIGN BASIS FLOODS AND PROBABILITY	X	X
B-31	DAM FAILURE MODEL	X	X
B-32	ICE EFFECTS ON SAFETY-RELATED WATER SUPPLIES	(REACTORS IN NORTHERN U.S.)	
B-33	DOSE ASSESSMENT METHODOLOGY	X	X
B-34	OCCUPATIONAL RADIATION EXPOSURE REDUCTION	X	X
B-35	CONFIRMATION OF APPENDIX I MODELS FOR "CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS"	X	X
B-36	DEVELOP DESIGN, TESTING AND MAINTENANCE CRITERIA FOR ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS FOR ENGINEERED SAFETY FEATURE SYSTEMS AND FOR NORMAL VENTILATION SYSTEMS	X	X

TABLE 2-2

(contd)

TASK NO:	TITLE:	RELEVANT TO BWR	RELEVANT TO PWR
B-37	CHEMICAL DISCHARGES TO RECEIVING WATERS	X	X
B-38	RECONNAISSANCE-LEVEL INVESTIGATIONS	X	X
B-39	TRANSMISSION LINES	X	X
B-40	EFFECTS OF POWER PLANT ENTRAINMENT ON PLANKTON	X	X
B-41	IMPACTS ON FISHERIES	X	X
B-42	SOCIOECONOMIC ENVIRONMENTAL IMPACTS	X	X
B-43	VALUE OF AERIAL PHOTOGRAPHS FOR SITE EVALUATION	X	X
B-44	FORECASTS OF GENERATING COSTS OF COAL AND NUCLEAR PLANTS	X	X
B-45	NEED FOR POWER - ENERGY CONSERVATION	X	X
B-46	COSTS OF ALTERNATIVES IN ENVIRONMENTAL DESIGN	X	X
B-47	IN-SERVICE INSPECTION OF SUPPORTS CLASS 1, 2, 3 AND MC COMPONENTS	X	X
B-48	BWR CONTROL ROD DRIVE MECHANICAL FAILURES	X	
B-49	IN-SERVICE INSPECTION CRITERIA AND CORROSION PREVENTION CRITERIA FOR CONTAINMENTS	X	X
B-50	POST-OPERATING BASIS EARTHQUAKE INSPECTION	X	X
B-51	ASSESSMENT OF IN-ELEASTIC ANALYSIS TECHNIQUES FOR EQUIPMENT AND COMPONENTS	X	X
B-52	FUEL ASSEMBLY SEISMIC AND LOCA RESPONSES	X	X
B-53	LOAD BREAK SWITCH	X	X
B-54	ICE CONDENSER CONTAINMENTS		X
B-55	IMPROVED RELIABILITY OF TARGET-ROCK SAFETY-RELIEF VALVES		X
B-56	DIESEL RELIABILITY	X	X
B-57	STATION BLACKOUT	X	X
B-58	PASSIVE MECHANICAL FAILURES	X	X
B-59	N-1 LOOP OPERATION IN BWR's AND PWR's	X	X
B-60	LOOSE PARTS MONITORING SYSTEMS	X	X
B-61	ANALYTICALLY DERIVED ALLOWABLE ECCS EQUIPMENT OUTAGE PERIODS	X	X
B-62	RE-EXAMINATION OF TECHNICAL BASES FOR ESTABLISHING SL's, LSS's, AND REACTOR PROTECTION SYSTEM TRIP FUNCTIONS	X	X
B-63	ISOLATION OF LOW PRESSURE SYSTEMS CONNECTED TO THE REACTOR COOLANT PRESSURE BOUNDARY	X	X
B-64	DECOMMISSIONING OF REACTORS	X	X
B-65	IODINE SPIKING	X	X
B-66	CONTROL ROOM INFILTRATION MEASUREMENTS	X	X
B-67	EFFLUENT AND PROCESS MONITORING INSTRUMENTATION	X	X
B-68	PUMP OVERSPEED DURING A LOCA	X	X
B-69	ECCS LEAKAGE EX-CONTAINMENT	X	X
B-70	POWER GRID FREQUENCY DEGRADATION AND EFFECT ON PRIMARY COOLANT PUMPS		X
B-71	INCIDENT RESPONSE	(NRC INTERNAL MANAGEMENT)	
B-72	HEALTH EFFECTS AND LIFE-SHORTENING FROM URANIUM AND COAL FUEL CYCLES	X	X
B-73	MONITORING FOR EXCESSIVE VIBRATION INSIDE THE REACTOR PRESSURE VESSEL	X	X

TABLE 2-2

CATEGORY C TECHNICAL ACTIVITIES

TASK NO.	TITLE:	RELEVANT TO BWR	RELEVANT TO PWR
C-1	ASSURANCE OF CONTINUOUS LONG-TERM INTEGRITY OF SEALS ON INSTRUMENTATION AND ELECTRICAL EQUIPMENT	X	X
C-2	STUDY OF CONTAINMENT DEPRESSURIZATION BY INADVERTENT SPRAY OPERATION TO DETERMINE ADEQUACY OF CONTAINMENT EXTERNAL DESIGN PRESSURE	X	X
C-3	INSULATION USAGE WITHIN CONTAINMENT	X	X
C-4	STATISTICAL METHODS FOR ECCS ANALYSIS	X	X
C-5	DECAY HEAT UPDATE	X	X
C-6	LOCA HEAT SOURCES	X	X
C-7	PWR SYSTEM PIPING		X
C-8	MAIN STEAM LINE LEAKAGE CONTROL SYSTEMS	X	
C-9	RHR HEAT EXCHANGER TUBE FAILURES	X	X
C-10	EFFECTIVE OPERATION OF CONTAINMENT SPRAYS IN A LOCA	X	X
C-11	ASSESSMENT OF FAILURE AND RELIABILITY OF PUMPS AND VALVES	X	X
C-12	PRIMARY SYSTEM VIBRATION ASSESSMENT		X
C-13	NON-RANDOM FAILURES	X	X
C-14	STORM SURGE MODEL FOR COASTAL SITES	X	X
C-15	NUREG REPORT FOR LIQUID TANK FAILURE ANALYSIS	X	X (ON COASTS)
C-16	ASSESSMENT OF AGRICULTURAL LAND IN RELATION TO POWER PLANT SITING AND COOLING SYSTEM SELECTION	X	X (FARM LAND)
C-17	INTERIM ACCEPTANCE CRITERIA FOR SOLIDIFICATION AGENTS FOR RADIOACTIVE SOLID WASTES	X	X

CATEGORY D TECHNICAL ACTIVITIES

D-1	ADVISABILITY OF A SEISMIC SCRAM	X	X
D-2	EMERGENCY CORE COOLING SYSTEM CAPABILITY FOR FUTURE PLANTS	X	X
D-3	CONTROL ROD DROP ACCIDENT	X	

TABLE 2-3

RISK-BASED CATEGORIZATION - NRC GENERIC ISSUES [12/]

- CATEGORY I: Potential High Risk Items

Group A:

TASK:	DESCRIPTION:
A-9	ATWS
A-6, A-7 A-8, A-39, B-55	Mark I, II programs, SRV pool dynamic loads and temperature limits, Target Rock Valve Reliability
A-17	Systems Interactions
A-40	Seismic Design Criteria
A-29	Design Features to Control Sabotage
A-10	BWR Nozzle Cracking
A-24	Qualification of Class IE Safety-Related Equipment

Group B:

D-57	Station Blackout Requirements
B-63	Isolation of Low Pressure Systems Connected to RCPB
B-30	Design Basis Floods and Probability
B-34	Occupational Radiation Exposure Reduction

Group C:

C-3	Insulation Usage Within Cantainment (sump blockage)

- CATEGORY II: Potential Low Risk Items

Group A:

A-3, A-4, A-5	W, B&W, CE Steam Generator Tube Integrity
A-1	Water Hammer
A-12	Fracture Toughness of SG/RCP Supports
A-2	Asymmetric Blowdown Loads
A-30	DC Power Supplies
A-15	Decontamination

Group B:

B-64	Decommissioning of Reactors (or parts of NSSS)

Not all items listed in the generic activities program pertain directly to reactor safety; indeed many relate to occupational risks and to relatively small publish risks from normal operation. For example, compilation of occupational radiation exposure reports from operating reactors has shown that annual exposures to station and contractor personnel has generally been increasing with time. At the present time, occupational exposures reported from U.S. operating nuclear facilities are averaging approximately 500 man-rem per reactor year, while the expected value for the annual accident exposure to the public associated with the plants analyzed in the WASH-1400 is predicted to be approximately 250 man-rem per reactor year. Thus, the occupational exposure exceeds the expected commitment associated with accidents at the present time. While a meaningful comparison of the risk associated with occupational exposure to the public risk associated with accidents is difficult, it should be recognized that the level of occupational exposures noted is actually being received. In contrast, the potential accident exposures are theoretical, being based on calculations involving releases of large quantities of radioactive material at low probabilities. 13/

In January, 1979, the NRC issued NUREG-0510 in which the classification of the over 130 generic technical issues

was once more revised. Seventeen issues consisting of 22 tasks were deemed as most important and henceforth were to be the "unresolved safety issues" (USI's) for the yearly Report to Congress. This list of 17 has since been expanded to a total of 26 issues.

The NRC has now reported resolution of 11 of the 26 USI's. A list of the remaining 15 issues is included as Table 2-4. 14/ However, even if all of the NRC's resolutions prove acceptable, implementation of the resolution on operating plants may not occur until the mid-1980's. In summary, it is difficult to escape the impression that the emphasis of the NRC on the generic technical activities programs has been to repeatedly reprioritize and relist the issues. List of generic issues have existed for years. Timely development of fixes for the fifteen USI's and implementation of hardware or procedural fixes would serve to eliminate a prime source of safety concern.

TABLE 2-4

UNRESOLVED SAFETY ISSUES

AND

APPLICABLE GENERIC TASK NUMERS [15/]

	UNRESOLVED SAFETY ISSUES:	NRC TASK NO.:
1.	Water Hammer	A-1
2.	Steam Generator Tube Integrity	A-3, A-4, A-5
3.	Mark I Long Term Program	A-7
4.	Reactor Vessel Materials Toughness	A-11
5.	Steam Generator & Reactor Coolant Pump Supports	A-12
6.	Systems Interactions	A-17
7.	SRV Pool Dynamic Loads	A-39
8.	Seismic Design Criteria	A-40
9.	Containment Emergency Sump	A-43
10.	Station Blackout	A-44
11.	Shutdown Decay Heat Removal Requirements	A-45
12.	Seismic Qualification of Equipment in Operating Plants	A-46
13.	Safety Implications of Control Systems	A-47
14.	Hydrogen Control Measures and Hydrogen Burns on Safety Equipment	A-48
15.	Pressurized Thermal Shock	A-49

2.4: SAFETY ISSUES RESULTING FROM TMI-2 ACCIDENT

The accident at TMI-2 sent shock waves through the entire nuclear industry as well as the NRC. The safety implications were grave reminders of the basic fallibility of human processes of design, regulation, maintenance and operation. It is not surprising, then, that there have been numerous reviews, commissions, inquiries and recommendations. Within the NRC there have been studies of feedwater transients by NRR[16], a study of the operational and radiological aspects of TMI by I&E [17], a "Lessons-Learned" report by DOR. [18] In addition, there has been an accident analysis by EPRI, [19] an extensive investigation by the President's Commission headed by Professor Kemeny and the NRC/TMI Special Inquiry Group under Mitchell Rogovin.

Each of these groups generated specific lists of faults and of recommendations. In some cases the recommendations are mainly to prevent the recurrence of a TMI-2 accident (e.g. vessel level measurement, AFW automation). However, some of these and many of the staff's other recommendations are generic which indicate areas of general safety concern. They are most comprehensively reported in the TMI Action Plan, NUREG-0737, and will not be repeated here. Among the generic items are the need to improve the man-machine interface, the need for better safety reporting/data evaluation, better simulators and training, more

analysis of the small break LOCAs and improved post-accident monitoring.

Industry reviews, such as the one conducted by EPRI and documented in NSAC-1, July, 1979 focused on the sequence of events. Solutions to problems identified appear to be an ongoing process through INPO and through owners groups.

2.5: SAFETY ISSUES WITH CONTINUING SIGNIFICANCE

In the previous chapters of this section we have listed hundreds of safety issues derived from various sources. The following are considered by MHB to be those which remain as items of safety concern for the present generation of reactors.

Unresolved Safety Issues

The fifteen USI's listed in Table 2-4 are certainly a first cut at prioritizing the several hundred candidate issues contained in the "generic issues" list and the TMI Action Plan. Satisfactory resolution including implementation at existing plants must be accomplished to bring the present reactor generation up to a minimum standard. Note: Several of the USI's are also listed separately. We assume the USI list will change with time so those of most significance have been separately discussed.

Human Factors

The control rooms of today's nuclear power plants are generally antiquated and poorly designed from a human factors

point of view. One person described them as based on 1950's concepts, designed in the '60's, built in the '70's, to be operated until the year 2000 or more. Each utility has design requirements which are unique for its systems and its control rooms. The result is vertical standardization by utility rather than horizontal standardization across the industry. This makes it difficult to standardize on a good human factors design and nearly impossible to have a simulator which closely resembles the many control rooms created for one generation of reactors. These deficiencies are clearly pointed out in the EPRI report, Human Factors Methods for Nuclear Control Room Design,[20] and are also mentioned to a lesser extent in the NUREG-0560 document.[21]

The TMI accident was originally laid at the feet of operator error. This blame was partially rescinded when it was discovered that the operator was working in a very difficult environment with less than desirable information. The difficulty of providing physical fixes to existing plants is a key factor in making this a particularly difficult problem to resolve.

Automation vs Manual Procedures

This issue could be included in the human factors concerns but is listed separately for completeness. The Feedwater Transient analysis report (NUREG-0560) recommended

that greater automation of some safety features be studied while at the same time it was recommended that the operator's actions be considered an extension of the defense-in-depth concept. This confusion on the balance between added automation and improved operator response provides inconsistency in the design guidelines. Dependence on operator response is limited at least at the point when the operator becomes overloaded with additional data and procedures. NUREG-0560 recommends that procedures be extended to cover multiple failures such as ATWS. The TMI operator was actually following a proper procedure for the plant readings but the wrong one to mitigate or prevent the accident. Proliferation of procedures may do more harm than good.

Another major question in this area is the value of the "10-minute rule" which says that during the first 10 minutes of a transient or accident, the plant should be designed so that no operator action is required for activating safety functions. Some countries (Sweden and Germany) have gone to a "30-minute rule." Criteria need to be developed to resolve these conflicts and appropriate actions taken at the existing plants.

Steam Generator Problems

PWR steam generator tube integrity has been identified as a generic problem for a number of years and in 1978 and continues to be listed as an "unresolved safety issue". Problems with steam generators in pressurized water reactors have resulted in substantial amounts of outage time and worker radiation exposure at most PWR's with several years of service. In the worst case situation, complete replacement of the steam generator tube bundles has been required in order to correct the severe problems of tube degradation. This has already occurred on Westinghouse designed units at Turkey Point in Florida and Surry in Virginia and is projected to be a future necessity at a number of other plants. The most recent newsworthy example of this problem occurred at the Ginna nuclear plant near Rochester, New York earlier this year. A recent issue of Inside NRC, an industry newsletter, said "Utilities are paying dearly for steam generator tube degradation." 22/ This article reports that NRC's assessment is that about 23% of non-refueling outage time is the fault of steam generator degradation, with very high cost to utilities merely in terms of replacing the lost power. Replacement of Virginia Electric and Power Company's Surry 1 and 2 steam generators cost the utility about $200 million and the ongoing replacement of Florida Power and Light's Turkey Point generators is estimated at $460 million. 23/ These expensive and lengthy

outages contribute heavily to the negative perception of nuclear plant reliability and safety. Improvements are being made in new steam generator designs to attempt to avoid some of the problems experienced in the past. Whether or not these changes resolve the problem remains to be seen.

Susceptibility of Primary System Components to Failure

One of the most serious failures a nuclear unit could sustain is a non-repairable defect in the reactor pressure vessel. Of somewhat less but still significant concern is the potential failure of portions of the large diameter piping systems making up the primary coolant system boundary. To date, no commercial U.S. plant has been forced to retire for the foregoing reason, but one commercial plant in Germany, the 237MW KRB unit has been permanently retired after approximately 10 years commercial service. One of the primary factors in prompting the retirement was the stress corrosion cracking failures experienced in the primary system recirculation loop piping.

The components most susceptible to this concern are:

(1) Failure of reactor pressure vessel due to extensive cracking of the structural materials. The most likely mode of such failure for BWR's appears to be fatigue cracking at major vessel nozzles, intertranular stress corrosion cracking of

cladding, safe-ends, CRD stub tubes, and other vessel internal components permanently welded in place.

(2) For pressurized water reactors, the current most serious problem is the issue of "pressurized thermal shock". The problem represented by PTS is a new concern expressed by the NRC into the degradation of PWR vessel material due to neutron embrittlement during operation of the unit. The material property changes make the vessel more susceptible to brittle fracture of the vessel during certain pressure and temperature transients that are occasionally experienced at PWR's and there is some indication that a number of the earlier PWR's may be forced to early retirement and/or extreme measures to restore the suitable characteristics of the pressure vessel.

(3) The problem of inter-granular stress corrosion cracking (IGSCC) of boiling water reactor primary piping systems has been a generic problem on BWR's for at least 15 years. This problem is currently listed by the NRC as technically resolved but continue to be the subject of extensive and expensive research programs throughout the industry. On past occasions, the NRC has formed special task forces to investigate the safety implications of this matter. These investigations are documented in NUREG-0531, Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants, February, 1979 and in NUREG-0691, Investigation and Evaluation of Cracking Incidents in Piping and Pressurized Water Reactors. Correction of pipe cracks at BWR's has had a significant affect on

plant reliability. The current replacement of recirculation system piping at Nine Mile Point will require at least one year of outage. A 237MW U.S. designed BWR in Germany was retired after only about 10 years commercial service primarily because of extensive IGSCC failures in the recirculation piping.

Some of the service sensitive piping has been designed out of the later BWR's but all reactors currently under construction will have recirculation loop piping with some susceptibility to IGSCC. With regard to the PWR pressurized thermal shock problem, all currently operating domestic Westinghouse reactor vessel can safely withstand PTS at the present time and the most vulnerable reactors are suitable for service to 1988. The investigation continues, however, and there have been several recent changes in the NRC policy and direction.

Anticipated Transient Without Scram (ATWS)

The ATWS event is potentially one of the most severe reactor accidents that could occur. It involves not only decay heat but full power heat as well and could lead to severe breaches of containment. The ATWS controversy (now 12+ years old) has centered around the probability (or

improbability) of the event.[24] From a safety standpoint, there is value in assuming it will occur and being prepared to mitigate the effects, or working to reduce the probability of occurrence. The NRC Staff estimated the chances of a severe ATWS accident at four-in-seven between now and the year 2010 [25] and subsequently reduced the estimate to a slightly lower probability (about a four-in-ten chance [26]).

The priority for resolution of this concern is mainly a result of the potential severity of the event if it should occur. The accident at TMI-2 has been declared a Class 9 accident by the NRC Staff. [27] This means it has exceeded the accidents analyzed in the licensing process and would have been classed as not credible prior to its occurrence. An ATWS accident has even more potential for destructive consequences, and therefore the public safety requires that it be prevented (or mitigated).

Reactor Siting Alternatives

The experience at Diablo Canyon nuclear plant has demonstrated the need for more restrictive and careful siting practices. It has cost years of delay and hundred of millions of dollars to modify the existing reactor plant to try to upgrade it for the larger earthquake now known to be possible at the site. The larger earthquake potential was not known at the time the site was approved.

The focus of the NRC has been to establish safety restrictions which, if complied with, will permit operation of nuclear reactors in potentially hazardous areas and sites. A solid review of all existing sites should be made and action taken on high risk sites, including phase-out if warranted.

Safety/Control Independence

The lack of independence of safety and control functions in some existing and planned reactors has contributed to concern on the part of the NRC and has even contributed to some accidents and transients, but still the plants are designed with shared or interrelated safety/control functions. For example, the HPI function at the TMI is used as make-up or control function but may also be called on as a safety function. The operator's action of turning it off manually should not have allowed its safety function to be compromised. Other reactor suppliers have also been identified as having weaknesses in their independence of safety and control functions. 28/ 29/ In several reactor accidents the reactor safety has been effected by a control system function, not a safety system. For example, Brown's Ferry relied on a condensate booster pump to provide cooling water, and TMI used the

non-safety block valves to stop the small LOCA through the PORVs. On the other hand, control functions can complicate safety actions. An example is the Integrated Control System (ICS) on B&W plants whose malfunction at the Rancho Seco plant caused a transient resulting in the steam-generators boiling dry. Criteria governing the independence of safety and control functions needs to be developed and implemented.

Water Level Measurement & ICC

Reactor vessel water level in a PWR is inferred from measurements of pressurizer level. Many of the post-TMI reviews have questioned this, but correction of this fault has not specifically required a direct method of measurement. Because of the difficulty of a level measurement based on differential pressure measurement in the core vicinity, some research is underway in the area of alternative techniques to level measurements, such as sonic, time-domain-reflectometer, etc. Similarly, the detection of inadequate core cooling (ICC) has been stated as a need but implementation of how to satisfy this need is still undecided.

Common-Cause Failures and the Single Failure Criteria

WASH-1400 made an attempt to quantify multiple failures due to a common cause. The effort has been criticized as being less than complete in that it would take an extraordinary

knowledge and insight to evaluate all the possible common-cause failures and their impact on safety. The events of TMI involved a series of failures which may be classed as common cause. This combination of problems is also described as the "Systems Interaction class of event. Is the single-failure-criteria adequate for licensing? Is the present limited consideration of common-cause failures adequate? Are extensive changes needed at the existing plants? A firm position is needed to supplement the present licensing criteria in these areas.

Radiation Level Buildup

The issue of radiation level buildup is a serious problem for current plants. To quote from an EPRI article on this subject:

> "As plants age, the long term trend is towards higher radiation dose rates at individual components, while the corresponding long term trend for regulation is for lower total absorbed doses. These two trends are inconsistent and something has to be done. EPRI has defined this problem, has served usefully in alerting the industry, and has obtained a fairly good quantification of what the problem is likely to be." 30/

The industry now appears to be aware of the problem of increasing radiation doses in the plants and the pressure towards reduced allowable exposure rates to the plant personnel and is taking action. Whether or not this problem proves to be correctable in the existing plants remains to be seen. Very little attention was given in the design and layout of existing

operating plants and plants now under construction. Maintenance operations exposure grows with plant age and with the number of plants and accounts for 80% or more of the total exposure. Is this level of exposure acceptable for the long term?

Future Safety Requirements

The TMI-2 accident demonstrated that serious reactor core damage accidents are a possibility and that consideration of core degradation and melting beyond the design basis accident should be conducted. Four of the tasks identified include research and analysis that could result in the imposition of significantly higher safety standards in the future that might significantly impact existing plants. These four tasks are:

(1) Research on phenomena associated with core degradation and fuel melting.

(2) Risk reduction for operating reactors for sites with high population densities.

(3) Analysis of hydrogen control.

(4) Rulemaking proceedings on degraded core accidents.

The research efforts associated with the above four tasks are under way and the degraded core rulemaking proceeding has been started. It is not likely, however, that the results will be known before another two to three years. The outcome of these investigations could have a significant impact on the futures of all existing plants. The possible outcome of the degraded core and

fuel melting investigation could be the requirement to add expensive mitigating systems such as filtered, vented containments, "core-catchers", or other exotic changes. A decision to "grandfather" existing plants would further degrade public confidence in their safety.

Plant Obsolescence

Safety Analysis Reports (SARs) are issued at the CP (preliminary or PSAR) and O.L. (Final or FSAR) stages to document the licensing basis for the plant. In the case of plants built in the late sixties and early seventies, the basis for licensing has changed appreciably with such changes as General Design Criteria, [31] Standard Review Plan for FSARs [32] and the addition of many new regulations (e.g., Regulatory Guides, Branch Technical Positions, and Industry Standards and Codes). Recognizing that for many older plants it would be difficult to assess their safety in terms comparable to a recently licensed plant, the NRC in 1978 started a program called the Systematic Evaluation Program (SEP). The goal of SEP was to reassess the safety adequacy of eleven operating reactors. Table 2-5 shows the eleven plants being reviewed. The eleven plants were chosen to represent older plants with a range of size, NSS vendors, and designs. Seven of the eleven are also operating under a Provisional Operating

License (P.O.L.). These POLs were issued for an eighteen-month period to provide an opportunity for the NRC to see how well the plant performed before issuing a Full Term License. In the case of the seven SEP plants, the POLs have now been extended for as much as fifteen years and have not yet been converted to F.T.L.

The plan is to review a list of 137 issues for each plant and document the safety status and compliance with today's regulations. A list of these issues is included in Appendix B. There is a possibility the NRC will also apply SEP to each of the remaining operating reactors. This would have been required by one proposed bill in Congress.[33/] Extension of SEP to all plants licensed prior to TMI is one possible way to gain confidence in the safety of existing plants.

TABLE 2-5

SEP PLANTS

SEP PLANT	DATE OF POL	TYPE OF LICENSE	TYPE OF NSS	NSS VENDOR	MWe
Dresden 1	1959	FTL	BWR	GE	200
Yankee Rowe	1960	FTL	PWR	W	175
San Onofre 1	1967	POL	PWR	W	430
Big Rock Pt.	1962	FTL	BWR	GE	67
La Crosse	1967	POL	BWR	AC	50
R.E. Ginna	1969	POL	PWR	W	490
Oyster Creek	1969	POL	BWR	GE	650
Conn. Yankee	1967	FTL	PWR	W	575
Dresden 2	1969	POL	BWR	GE	794
Millstone 1	1970	POL	BWR	GE	680
Pallisades	1971	POL	PWR	CE	805

2.6: SAFETY/RELIABILITY CONCLUSIONS

The preceding subsections have discussed safety and reliability concerns arising from the experience of the past. All of these issues must be dealt with in some way if the public confidence is to be established. The continual reordering of the various priority levels does not make the issues go away. Implementation of corrective action for those concerns of importance would help to reassure the public that those issues judged insignificant are in fact benign. From the many issues listed, we conclude that the following are most significant:

- Resolution of unresolved safety issues.
- Solution to steam generator problems.
- Corrective action for primary system failure modes (PTS, pipe cracks, & internal component cracking).
- Finalization of ATWS fix.
- Improved PWR water level measurement and assurance of inadequate core cooling indication.
- Reduction of radiation level buildup.
- Decisions regarding core damage mitigation.
- Improvements in human factors.
- Further definition of automatic safety system requirements.

- Improved reactor siting criteria.
- Improved separation and independence of safety and control functions.
- Improved criteria for common-cause and single-failure design.
- Adequate assessment and response to aging and obsolescense problems.

With regard to the overall assessment of reactor safety, we conclude that adequate attention has not been given to safety issues, or at least, not to the proper safety issues. It is possible that the level of safety presently attained is adequate, but that has not been demonstrated and the public is not convinced that is the case.

Reactor plant reliability as judged by plant availability and capacity factors is presently below expectations. A discussion of plant performance as it correlates to size is contained in Section 5, and a summary of recent reactor performance is contained in Appendix D. Capacity factors of 80-85% were used in most economic evaluations for plant justification. A factor of about 60% is now the norm.

SECTION 2

LIST OF REFERENCES

1/ NUREG-0410, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, January, 1978, U.S. Nuclear Regulatory Commission, Washington, D.C., pages D-1 to D-7. (Note: NUREG-0371, entitled Approved Task Action Plans for Category A Generic Activities, Vol. 1, Rev. 1, is contained as Appendix F of NUREG-0410.)

2/ "Investigation of Charges Related to Nuclear Reactor Safety," Hearings Before the Joint Committee on Atomic Energy, Vol. 2, 1976, U.S. Government Printing Office, Washington, D.C., pages 1200 to 1445.

3/ Letter, M. Bender, ACRS Chairman to Joseph Hendrie, NRC Chairman, entitled, "Status of Generic Items Related to Light-Water Reactors: Report No. 6," dated November 15, 1977. (An updated report was issued March 21, 1979, Further updating has routinely occurred.)

4/ NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRC to NRC Staff, December, 1976, U.S. Nuclear Regulatory Commission, Washington, D.C.

5/ NUREG-0153, Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976, Memorandum from Director, NRC to NRC Staff, December, 1976, U.S. Nuclear Regulatory Commission, Washington, D.C.

6/ Ibid 3.

7/ Ibid 2, page 1203.

8/ Ibid 1, page 7.

9/ Ibid 1, page 7.

10/ NUREG-0510, Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, Report to Congress, U.S. Nuclear Regulatory Commission, January 1979.

11/ Taylor, et al, Summary Report on a Risk Based Categorization of NRC Technical and Generic Issues, preliminary draft issued by U.S. Nuclear Regulatory Commission, Washington, D.C., page 1.

12/ Ibid 11, page 2.

13/ Ibid 11, page I-23.

14/ NUREG-0606, Vol. 4, No. 3, Unresolved Safety Issues, August 20, 1982.

15/ Ibid 14.

16/ NUREG-0560 - Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by B&W Company, U.S. NRC, May, 1979.

17/ NUREG-600 - Investigation into the March 28, 1979, Three Mile Island Accident by Office of Inspection and Enforcement, August, 1979.

18/ NUREG-0578 - TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations, U.S. NRC, July, 1979.

19/ NSAC-1 - Analysis of Three Mile Island-Unit 2 Accident, EPRI, Nuclear Safety Analysis Center, July, 1979.

20/ Ibid 18, pages A-9 and A-19.

21/ Ibid 19, Introduction (page unnumbered).

22/ Inside NRC, McGraw-Hill Publications.

23/ Ibid 22.

24/ Ibid 17, pages I-B-1 through I-B-12 and pages II-F-1 through II-F-13.

25/ Human Factors Methods for Nuclear Control Room Design, EPRI NP-1118-SY, June, 1979, Summary Report.

26/ Ibid 25, page 8-7 through 8-9.

27/ NUREG-0460 - Anticipated Transients Without Scram for Light Water Reactors, U.S. NRC, Vol. I and Vol. II, April, 1978, Vol. III, December, 1978.

28/ Ibid 27, Vol. I, page 92.

29/ Ibid 27, Vol. III, page F-6 and Appendix F, Figure 1 shows a total probability of core melt or excessive pressure reaching about 2×10^{-2}/yr in 1990. For the period between 1990-2010 this would be about a 4 in 10 chance.

30/ EPRI Journal, Jan/Feb 1979, page 17.

31/ 10 CFR 50, Appendix A.

32/ NUREG-75/087, November, 1975.

33/ The Bingham Amendment to the House Appropriations Bill would have required a safety assessment of all operating reactors.

SECTION 3

ASSESSMENT OF INDUSTRY APPROACH TO REACTOR PLANT MANAGEMENT

3.1: INTRODUCTION

A key factor for the future of nuclear power is the capability of the total nuclear establishment to effectively manage and direct the business of designing, building, operating, and regulating nuclear plants. Past experience at TMI as well as current experience at South Texas, Zimmer, and Diablo Canyon indicate that substantial improvement is needed in the existing structure. A second area of concern, and one that has been receiving much attention of late, is the likelihood of degradation or loss of the technical capability to manage such an enterprise in the future, given the fact of the dramatic decline in the number of units ordered.

Another aspect of this complex question is whether or not such a broad and demanding operation can be effectively managed without the strong central control inherently lacking in our democratic society. Admiral Rickover's testimony to Congress in the TMI aftermath attributed much of the success of the Navy nuclear program to the strong central control over all aspects of the operation, ranging from initial design to

details of propulsion unit operation. This degree of control does not exist in the commercial nuclear program and it is unlikely that it ever could exist in a system of hundreds or thousands of privately-owned companies. The issue to be addressed then is whether the level of effort devoted to the broad aspects of nuclear plant management is high enough and the quality control stringent enough to ensure that the risk is low enough to satisfy the minimum and perceived needs of this industry.

3.2: SAFETY ASSURANCE EFFECTIVENESS

We use the term Safety Assurance Effectiveness to encompass the essential combined structure of safety-related philosophical, developmental concepts, administrative, and procedural methods and processes which must be effectively performed to assure that the total nuclear establishment actually produces facilities that meet the desired safety goals.* Quality assurance and the issue of regulatory effectiveness are two well-known parts of this structure which have received much attention and discussion, but which continue to elude acceptable realization.

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*For further discussion on safety assurance effectiveness, the interested reader is referred to the MHB report for the Department of Energy, ALO-62, entitled Improving the Safety of LWR Power Plants, printed in April 1980, and available from N.T.I.S.

There are many additional aspects of this safety assurance structure, any one of which can (and does) break down, resulting in degradation of the safety actually achieved. The breakdown can occur throughout a wide range of the lengthy process required to produce and operate a nuclear plant, ranging from a failure of the basic design criteria, to the lack of quality in the construction process, and on to the relatively expected, but unpredictable, factor of human errors committed under abnormal or emergency operating conditions.

To further complicate the matter is a complex array of regulations, organizations, interfaces, and divided responsibilities that is continuously changing throughout the life of the nuclear project as it moves from the conceptual stage, into design and construction, and, finally, into operating status. Not only does the nature of the project and its needs change, the organizations and personnel within a given institution change from phase to phase. There are four basic phases to consider:

a. Design

b. Licensing

c. Construction

d. Operation

Within the three major entities involved, the vendors, the utility, and the regulatory agency, organizations are

generally emplaced to implement the specific activities of each of these phases. For example, the typical nuclear steam supply system vendor will have a separate design group, a licensing operation, a project (construction) organization, and, finally, an operating liaison group. Each one of these groups deals with a (similar) counterpart in the utility or the NRC. Voluminous communications, procedures, and documentation are required to establish and define the various interfaces. With the divided responsibilities, periodic transfers of management, and the changing emphasis for the project needs, there are many opportunities for errors or less than optimum attention, and some functions may be overlooked or not completed.

The difficulty of managing all aspects of this complex situation makes time the enemy. Difficult confirmatory programs or regulatory decisions are delayed, but the concrete continues to pour per the construction schedule. Prime examples of this process of indecision are the ten-year plus consideration of what to require for the resolution of ATWS, or the LOFT program that began about 15 years ago but which is incomplete at this date. The technical complexities of many such issues, coupled with the "impenetrable shield of middle management" present not only in the NRC, as discovered by the Kemeny Commission, but prevalent throughout

an industry made up primarily of large organizations, acts to make timely resolution of such problems nearly impossible.

All of these important issues might be considered to be theoretically covered by the quality assurance programs now required by the NRC's 10 CFR 50 Appendix B. The 18 QA criteria contained in Appendix B basically cover all of the safety assurance requirements, and if the 18 criteria were successfully carried out in a timely manner, they would probably provide satisfactory implementation of the safety program. The problem is that many of today's reactors were built without benefit of Appendix B, and many others are lacking in effective implementation of the criteria.

The importance of satisfactorily addressing these deficiencies cannot be overemphasized. It may do little, or absolutely no good, to spend hundreds of millions of dollars and countless resources to develop and test the best nuclear plant safety systems in the world if they can be bypassed by an improperly qualified operator, or defeated by a forgotten procedure. Light water reactor safety improvements in probability reduction and in consequence mitigation are meaningless if those improvements are not actually achieved in the plant hardware and systems. Because these problems are for the most part human failings, or at least produced

as a result of human omissions or commissions, it is more difficult to describe "fixes" for them. Following, however, is a listing of specific problems and some suggestions as to how to minimize or correct some of them.

Inherent Design Problems

As discussed above, much of the difficulty inherent in the design process for nuclear reactors stems from the long time cycles involved in the design and construction of such a plant and also from the split responsibilities assigned to the various companies and organizations involved. Aside from these ongoing problems of design management control, the following are problem areas unique to the nuclear industry:

1. Uniqueness Versus Standardization - The safety value of standardization has been extensively discussed and it is generally agreed that standardization is a positive factor. However, the unique practices of any of the private utilities tends to defeat the standardization benefit. Changes in plant layout and configuration, location of key pumps and valves, and control room configuration are a few examples of this uniqueness.

2. Design Verification - Appendix B of the Quality Assurance Criteria requires that appropriate design verification be performed. The way in which it is to be done is relatively loosely defined and in many cases a second level analytical review is substituted for system or hardware tests. The size and complexity of nuclear plants is the largest single factor in forcing such choices.

3. Conflicting Design Goals - There are conflicting design goals which tend to make optimal design nearly unachievable. These conflicts exist both in the technical and commercial domains. An example of a technical conflict exists in the location of primary containment isolation valves. The NRC General Design Criteria require the valves to be placed as close to the containment boundary as practical. Some of these valves are also utilized for operation or protection of external systems and/or pressure vessels and are required by the ASME Code to be as close to the equipment as practical. Both goals cannot be satisfied. This is one of many such technical conflicts.

An example of a commercial conflict is the scope definition conflict between the nuclear steam system supplier and the architect engineer. NSSS suppliers normally operate on a fixed price plus adders for scope increase. Architect engineers normally operate under contracts with cost plus arrangements. As a result, the interface is constantly changing and there are many opportunities for mis-management and mis-communication.

4. Design Completion - Many aspects of nuclear plant design are totally unsettled when plant construction begins and many aspects of the design may still be incomplete when plant operation starts. One-step licensing of nuclear plants has been proposed as an improvement to the licensing process. This could help to force a more complete definition of design prior to construction but that is not assured. Recent experience at Zimmer and at Diablo Canyon would lead one to believe that an extensive independent design verification review might be a valuable feature to

require at every plant prior to initial operation.

5. Loss of Continuity - The Quality Assurance Criteria specify that changes made to plant or systems are to be reviewed by the original designer. With architect or supplier contractual responsibilities terminating at the completion of construction this requirement is sometimes difficult to meet. If an ongoing relationship is not contemplated, it is essential that the original design goals and guidelines be thoroughly documented and continuously updated. Failure in this area can have significant impact on the maintenance of safety objectives throughout the life of the plant.

Plant Construction

Quality control of nuclear plant construction program has received much attention in the past year and Chairman Palladino of the NRC has been somewhat outspoken on his surprise that such quality discrepancies continue to exist. It is probably inherent in the construction of a nuclear

plant that quality problems will be a serious problem. Following are a few factors we consider to be key in this industry:

1. Availability of Qualified People - Nuclear plants for the most part are situated in remote locations (as they should be) due to the safety requirements of plant siting. This, however, provides difficulty in that the pool of qualified workers may be smaller so that the best location for safety is the worst location for construction resources.
2. Lengthy Construction Cycles - The 8 and 10 year long construction cycles being experienced at nuclear plants means that a significant turnover of personnel occurs and unless the documentation is maintained in an up-to-date fashion, direct knowledge of the status of many of the key systems and components may become scattered or lost. Changes become so numerous and/or extensive that the original drawings become hard to decipher and piecemeal descriptions of systems are necessary in order to keep up-to-date with changes.

3. Logistics - The logistics of the management of thousands of construction workers in the relatively compact space making up a nuclear plant becomes a nightmare of scheduling and of maintaining quality control. The current investigations at Zimmer, Diablo Canyon, and at the South Texas project give vivid examples of this issue.

Operational Quality

The application of rigid quality standards to operating nuclear plants is relatively new. Utilities have had some difficulty in adopting these standards or at least in applying them effectively. This may be partially due to the long time utility guideline that the most important thing to be done is to "keep the lights on". This may be exactly counter productive to the goal of operating a plant in the safest, most reasonable way. Following are some factors that in our opinion serve to degrade the quality of plant operation:

1. The Learning Curve - Operating a plant with the discipline required to satisfy the safety and quality assurance needs has generally not been followed in non-nuclear plant operations. Accordingly, utilities must "learn the ropes"

and it generally takes more than one plant's worth of experience to become effective.

2. Rising Costs - The difficult financial times being faced by most utilities serves to work against the updating and safety improvement of existing operating plants. Money is difficult to obtain and the money that is available to the utilities for the most part has already been committed to supporting the extensive construction cycles. As a result, plant improvements are often limited to those that are absolutely essential or to those that are directly imposed by the regulatory body. As a result, reliability and/or maintainability receives a lower priority and quality of the operation suffers.

Regulation

Improvement of the regulatory process has also received much attention of late and is discussed further in later sections of this report. However, for the purpose of this section, it is appropriate to point out a number of factors which in our view have caused the regulatory effort to be much less effective than desirable. Of significance are:

1. Timeliness - It is extremely difficult to implement modification programs at nuclear plants, particularly if extensive and basic reanalysis is required. It is relatively common for the NRC to approve waivers for certain plants so as to permit operation while investigation is underway. A good example of this problem is the GE BWR Mark I containment reassessment program which resulted in operation of approximately 20 plants for several years under a waivered condition. This problem was first identified in 1975 and is yet to be fully corrected at all of the affected plants. The problem is how to make an intelligent risk-benefit decision with the large economic pressure that can be present if a generic issue affects a substantial number of the generating plants.
2. Owners Groups - A common response of utilities to generic regulatory issues is to form an owners group so as to permit assessment of the problem on a generic basis. The NRC has encouraged such actions as it lightens the review workload and speeds the process.

However, it must be recognized that the lineup of ten or more utility representatives against the desires of a single NRC technical staff individual may be somewhat intimidating. This fact must be recognized by the NRC and appropriate management attention given to such situations.

3. Cost Benefit Analysis - Analysis of problems as an aid to decision making is a desirable feature and has been receiving more attention, particularly with regard to probabilistic risk assessment and cost-benefit analysis for the making of decisions regarding plant modifications. Safety improvements can be rejected or justified depending upon how the analysis is performed and how the parameters are valued. For example, ALARA decisions are often based on an outdated value of $1,000 per man/rem and as a result few changes can be justified. Similar care must be exercised for PRA and other cost benefit analysis. The potential misuse of analysis in regulation is a significant concern.

3.3: CONCLUSIONS

In general, we find that the industry's approach to the design, construction, operation, and regulation of nuclear power plants has not yet reached the level of effort adequate to bring the risk of nuclear power to an acceptably low level. This is of course a subjective conclusion and we are well aware that numerical "analysis" such as PRA may lead to the opposite conclusion, depending upon attribute valuation and acceptance criteria.

Implementation of the changes we see needed to ensure the desired improvement in safety assurance effectiveness is itself impeded by the same factors that impede the effectiveness of the whole regulatory process. These impediments are bureaucratic inertia, divided responsibilities, and a reluctance to make changes to existing plants. The quality assurance program required by 10 CFR 50 Appendix B would basically correct the identified concern if the program were to be carried out in a thorough and all-reaching manner. This does not appear to be being done and there are some reasons to believe that the improvements that should have been gained through the broad application of Appendix B have not in fact been achieved. As expressed in the statement of Richard B. Hubbard before the House Subcommittee on Energy and the Environment [1/], the QA

program in its simplest form requires only that you "plan what you do, do what you plan, and document what you did". A copy of Hubbard's statement is attached as Appendix C. It describes some examples of QA system breakdowns that have occurred.

A recent common approach to quality system breakdown has been to retain an independent technical group to perform a design or quality assessment of projects where serious breakdowns have occurred. This is currently underway in the Teledyne assessment of Diablo Canyon, a recent C. F. Braun assessment at LaSalle, a Quadrex review at South Texas and is currently being planned for the Zimmer plant. We believe that this is a positive step towards the effective assurance of reactor safety and is one that may be considered to be generically applied for future plants. It is quite likely that an effective independent assessment program carried out approximately one year prior to fuel loading could easily more than pay for its cost in terms of avoided future outage time. In our experience in the nuclear industry, a substantial amount of operational outage time is caused by allowing the plant to begin preliminary power operation before all necessary design reviews and plant modifications or completions have been performed. Premature start-up adds substantially to the time required for completion

since work on a radioactive system is significantly more complex and time consuming. We recommend that a much broader use of independent assessment be conducted prior to the time that trouble is normally identified, i.e., far enough in advance of the award of the operating license so that corrective action can be taken in a timely fashion.

Operational QA plans would also benefit from periodic independent assessment, followed by in depth NRC review. Present NRC staffing does not provide for this level of review. We also believe that the NRC QA plan should receive similar independent scrutiny and public review. Perhaps INPO could perform an operational effectiveness audit that would help to focus the regulatory resources.

SECTION 3

LIST OF REFERENCES

1/ Statement of Richard B. Hubbard Before House Subcommittee on Energy and the Environment, November 19, 1981.

SECTION 4

ALTERNATIVE PROPOSALS

4.1: INTRODUCTION

The purpose of this section of the report is to comment on a few of the energy supply alternatives that have been proposed for the continuation of the current generation of reactor plants. We are generally limiting this discussion to a review of the alternates described in the "Survey and Assessment of the Technological Options" report authored by A. P. Fraas.[1/] It should be noted however, that the only alternative discussed therein are variations of the nuclear option. The viability of nuclear power as an energy supply option will ultimately depend on a balancing of energy demands and utilization of all nuclear and non-nuclear options. In our opinion, it is likely that the lessening of energy demand, coupled with a broader application of conservation techniques and the development of some of the "soft technologies" may serve to reduce the nuclear option need further. We recognize that the purpose of this study is intended to focus on nuclear power, but we would be amiss if we did not call attention to the continuing need in the development of other energy options.

The Fraas report evalutes the following alternative nuclear reactors:

Next generation light water reactors (LWRs).

Heavy water reactors similar to the CANDU (HWRs).

High temperature gas-cooled reactors (HTGRs).

Process inherent ultimately safe reactors (PIUS).

Our comments on factors related to these options are presented in the following subsections.

4.2: ASSESSMENT OF ALTERNATIVES

The scope of our review did not include allowance for a significant evaluation of nuclear power alternatives nor was time available for an exhaustive review of the Fraas report. We have some familiarity with the four alternatives assessed and we did conduct a rapid review of the Fraas report.

Advanced LWR Designs

A portion of the Fraas report discussed the post-TMI improvements in conventional LWRs and also discussed design evolution of the U.S. LWR models, including the SNUPPS PWR, the GE BWR/6 and European versions of the U.S. PWRs. Operating experience with these reactor models is limited or non-existent but some relevant facts can be stated.

1. None of the SNUPPS PWRs are yet in service so the efforts stated of improving safety and reliability have yet to be demonstrated.
2. The UK-CEGB version of the SNUPPS design (Sizewell B) has yet to be committed. It is our understanding that the CEGB has preliminarily concluded that the improved safety Sizewell configuration is too expensive and consideration is being given to building Sizewell as a "present generation" type PWR.
3. The BWR/6 containment system improvement identified by Fraas which incorporates two containment envelopes and a suppression pool of water has in fact been utilized in some form in all BWRs built in the United States since 1963. There is substantial reason to believe that the BWR/6 version of pressure suppression will result in higher radiation exposures to workers for operational transients and certain classes of accidents than the earlier BWR models.
4. The safety improvements attributed to the KWU reactors with regard to containment size,

separation design, base mat resistance to core melt, etc. would seem to represent significant improvements over present generation reactors. These would be particularly desirable features for future reactors to be built in the U.S. but could not likely be backfitted to any significant degree.

Heavy Water Reactors

Relevant experience with heavy water reactors is primarily limited to that utilizing the Canadian heavy water design (CANDU). The Fraas report points out the apparent paradox of the CANDU reactor being the most complex of the reactors assessed but yet having the highest availability. He concludes that the stringent requirements dictated by this complex design may have demanded such high quality that it has been given such careful attention so as to produce high availability as a side benefit. This is a lesson that can and should be applied to the LWRs; quality doesn't cost, it pays. In further evaluation of the heavy water reactor for future application in the United States, we believe the following are significant factors to consider:

1. There is no HWR supplier in the United States and the commercial base for producting it does not exist.

2. Licensing of the HWR in the United States would require the development of a regulatory capability that is presently non-existent.
3. The capital costs of this reactor is no doubt substantially higher than the LWR and might experience further escalation in translating it to a U.S. licensable design.
4. It is not obvious why this reactor design would be perceived by the general public as being more acceptable than the current LWR design.

High Temperature Gas Cooled Reactors

One commercial HTGR has been licensed and is in operation in the United States (Fort St. Vrain). While this reactor is substantially smaller than most LWR models, it does represent a commercially applicable licensable design and is one for which a U.S. supplier and technological base exists. Following are significant factors to consider in assessing the acceptability of the HTGR:

1. The technological base for designing, licensing, and constructing this reactor does exist.

2. The one existing HTGR has not demonstrated a high degree of commercial reliability. This may or may not be evidence of a design weakness in this area.
3. The Fraas report points out that the annual exposure at the HTGR experienced at Fort St. Vrain is less than 1/10 of that seen at the LWRs. Since only one such reactor exists, this may not be representative of the exposure expected for a larger population of HTGRs.
4. The incomplete fuel cycle of the HTGR represents a cost and safety unknown that must be given further consideration.
5. The capital costs of an HTGR is admittedly higher than LWRs and sufficient experience may not yet exist to ensure that this is properly quantified.

Process Inherent Ultimately Safe Reactor (PIUS)

The design concept incorporated in the PIUS reactor appears to have many promising and desirable features. The fact that this reactor has not yet been built nor the operability demonstrated is the most significant single factor acting as a disadvantage to its use. Following are other significant factors:

1. A U.S. supplier of this design does not yet exist. However, in view of the relatively early stages of development it is assumed that a U.S. supplier could be found.
2. The single biggest technical concern with regard to operation of this plant would appear to be the potential difficulty of performing maintenance and inspection on the components contained within the primary boundaries.
3. Exportation of this "safe" reactor to less technically sophisticated countries would still represent a potential proliferation concern.
4. The unproven costs associated with this reactor represent a large uncertainty in its future viability.

4.3: CONCLUSIONS

We conclude that there are nuclear alternatives to the LWR that can and should represent safety and reliability improvements and which may be more acceptable to the U.S. public. None of these designs however would appear to change several aspects of the public's concern regarding nuclear power

namely, radioactive waste disposal and nuclear weapons proliferation concern. It is conceivable that the PIUS reactor could be "marketed" to the public and be perceived as being so much safer that these concerns might be overcome. The other reactor alternatives do not appear to have any substantial edge over present LWR's. The PIUS reactor would appear to be many years away from commercialization and would require a large commitment of resources, probably at the federal level, to bring it to a viable design.

SECTION 4

LIST OF REFERENCES

1/ "Survey and Assessment of the Technological Options Available to the Nuclear Industry in the 1980 to 2000 Period", A. P. Fraas, November 1982, Institute For Energy Analysis, Oakridge Associated Universities.

SECTION 5

REACTOR SIZE

5.1: INTRODUCTION

The question of reactor size has been raised as a possible issue potentially affecting the safety and reliability of nuclear units. The opinion has been expressed by some that a smaller reactor unit might be easier to design, construct, and operate and that the smaller units therefore should be promoted. This section of this report shall offer some comments and observations on this issue and will also present an analysis of performance of existing reactors during the last several years.

5.2: QUALITATIVE ASSESSMENT

A fairly extensive discussion of the different size dependent characteristics of nuclear plants has been prepared for the OTA and presented in a November 19, 1982 report by ENSA. 1/ We will not reiterate here all of the facts in that report but will comment only on several that seem to be of more importance. The following factors seem to be of most significance:

1. Safety Considerations - There would seem to be no logical technical basis for concluding that the probability of an accident is greater for a larger than for a smaller plant. Both plants used the same basic safety system configuration and accident initiating events should occur with approximately the same frequency. Therefore, if training, testing, maintenance, and other human factors are controlled to the same degree at both large and small plants, there should be no difference in the probability of accidents related to plant size.

   With regard to accident consequence, the ENSA report correctly observes that the worst-case accident consequences are approximately proportional to the size of the reactor core. There is, therefore, a direct size dependency relationship for the consequences of a serious reactor accident. The total risk per KW installed would tend to be equal for both large and small since the lower consequence of a small reactor would be coupled with a larger number of plants.

We conclude that while there may be no significant technical basis for stating that a smaller plant has a safety advantage, it should be easier to develop effective accident response plans if the area potentially affected is smaller. Fewer members of the general public would be involved in preparation of such plans and fewer local agencies would be required to coordinate plans. A smaller plant would likely be perceived as affecting fewer members of the public and would probably be more readily accepted.

2. Cost - One of the most significant contributors to the large costs being experienced at nuclear power plants today are the indirect cost factors including project management and interest during construction. It is reasonable to assume that a large plant, requiring more material and more human resources, will require a longer time to construct than a smaller plant. The result would be a tendency for a larger proportion of indirect costs for a larger plant due to the longer schedule. This would be compensated for, on the other hand, by a larger proportion of

fixed costs due to project management, licensing, etc. Our judgement is, however, that there is an optimum size and that it is in the 600-800 MW range.

3. Institutional - There are pros and cons for size related to the plant operating staff. The Swedish nuclear establishment feels that operating from a relatively small body of personnel allows for an "espirit de corps" and a pride of competence that tends to improve performance. A similar attribute is described in the Navy reactor program. The counter argument of this concept is the possibility of lacking technical depth. We tend to accept the "smaller = excellence" theory based on the morale and general atmosphere found in smaller plants we have visited.

4. Philosophy of Acceptability - We are not aware of any studies that have been conducted to determine if the public would be more likely to view a smaller project with less suspicion. Our intuition tells us that would be the case. People tend to be troubled by the feeling that they have no control over outside events that may affect their lives. A smaller plant may ease that concern. We have

also observed that differences of opinion tend to be less sharply defined if there is an opportunity for two-way communication. A smaller plant and a smaller operational structure would tend to make that more likely to occur. It is unlikely, however, that a nuclear plant could even be small enough to bring it to the "mom and pop" grocery store status. If it did, "familiarity" might "breed contempt".

5.3: ANALYSIS OF PERFORMANCE FACTORS

We have examined the recent performance of large and "small" (less than 771 MWe) BWRs and PWRs to determine if any significant differences exist. We have used data for the period of January 1, 1979 through May 31, 1982 as reported in the NRC's Greybook. [2/] A summary of this, by plant, is contained in Appendix D. Overall capacity factors for groups of plants can be calculated by weighting each plant's capacity factor by the plant capacity. Thus:

$$CF = \frac{\sum_{i=1}^{n} cf_i \times \text{capacity:}}{C}, \text{ where}$$

CF = the overall capacity factor for a group of plants
cf_i - the individual capacity factor for a plant in the group
capacity: = the maximum dependable capacity for a plant in the group
n = the number of plants in the group
C = the sum of the capacities of all the plants in the group

The same formula may also be used to calculate the forced outage rate for a group of plants, if forced outage rates are substituted for the capacity factors.

Comparison of BWR and PWR Performance

Using the above formula, overall performance factors were calculated for BWR's and PWR's, as shown in the following table:

Group Performance Factors, BWR vs. PWR

Factor	All BWR's	All PWR'S
Capacity factor	61.9	59.6
Availability factor	70.4	67.3
Forced Outage Rate	10.5	16.7
Number of Plants	22	45

Although BWR's have slightly higher performance statistics, the difference is not large enough to support a conclusion that the BWR's are more reliable.

Large Plants vs Small Plants

The BWR and PWR groups were divided into "large" and "small" plants, based on the median capacity in each group (i.e., the median BWR capacity was 771 MW; any plant with a higher capacity than 771 was "large"). The capacity factors and forced outage rates calculated are shown below.

Group Performance Factors, Large vs. Small Plants

Factor	BWR's		PWR's	
	Large	Small	Large	Small
Capacity Factor	60.7	63.6	56.8	62.4
Forced Outage Rate	11.2	9.5	18.4	14.3

The small plants consistently appear to have a slightly better average performance. This may not be statistically significant since it may relate to plant age. This is discussed in the following paragraphs.

Older Plants vs. Newer Plants

Within both the PWR and BWR groupings, plants were subdivided into "older" and "newer" groups by relation to

the median plant age in each group. The results of comparing performance factors for older and newer plants are shown below.

Group Performance Factors, Older vs. Newer Plants

Factor	BWR's		PWR's	
	Older	Newer	Older	Newer
Capacity Factor	64.3	60.1	61.2	58.6
Forced Outage Rate	8.7	11.9	15.4	17.5

The older plants have slightly higher capacity factors as a group than the younger plants. As in the large vs. small comparison, the difference is not very great but it is consistent. This similarity of results is to be expected, since the older vs. newer grouping corresponds roughly to the small vs. large grouping.

5.4: CONCLUSION

It may be concluded that small plants have performed more reliably than large plants, and there are subjective reasons that may be used to explain this. There appears to be no overwhelming technical reasons that would support the theory that small plants are inherently safer except for the smaller potential for accident consequences. We conclude that

a careful study should be performed to determine optimum plant size, using cost data expected throughout the life of the plant. It also seems logical that smaller plants would be more readily accepted by the public.

SECTION 5

LIST OF REFERENCES

1/ Nuclear Power Plant Size (Draft), Prepared by D. A. Prelewicz, J. B. Keating, and M. N. Haass, November 19, 1982, ENSA Inc.

2/ NUREG-0020, Data For Operating Plants as of January, 1979 and May 31, 1982.

SECTION 6

LICENSING PROCESS

6.1: INTRODUCTION

This section of the report provides our comments on the effectiveness of the current licensing process. It also comments on some of the changes in licensing that have been proposed. Included in our comments is a brief review of certain aspects of reactor regulation contained in the SC&A report prepared for OTA. 1/

6.2: LICENSING EFFECTIVENESS AND DEFICIENCIES

The licensing process has been blamed for much of the delay in plant construction cycles and for a substantial share of the cost overruns experienced at most nuclear projects. It is not clear that this blame is realistically founded but the regulation is a logical whipping boy for all such negative influences. It is undoubtedly true that the nuclear plant licensing process has a significant effect on the schedule and cost of a project. That this is all undesirable has not been demonstrated. The positive aspects of licensing are substantial, both in public health and safety and in the protection of the large financial investment represented by a nuclear facility. Witness the losses incurred when effective regulation is not achieved as at TMI.

The point can be made that while nuclear regulation has not been as effective as it could be or should be, it has at least been effective enough to date to have avoided catastrophic (lethal) accidents. Following are listed some of the more significant licensing deficiencies observed during our participation in the process.

1. Timing - There appears to be no good time to conduct a licensing review. The review at the CP stage is too early for significant detail to be available. The OL review is too late to have a significant impact unless gross deficiencies are discovered.
2. Documentation of Review - The licensing review conducted by the NRC is reported in the SER in legalistic and "cookbook" fashion stating primarily that the appropriate regulations have been met. There is no way for a third party to follow the design trail short of going to hearing so as to obtain discovery rights. Even then the trail is obscured by generic references and proprietary shields. If public confidence is to be developed, a transparent review process must be available.

3. Intent vs. The Letter of The Regulation - The various regulations for the most part are based in substantial technical fact. However, in translating that base into a legal process, the intent sometimes is lost and a rigid adjudicatory process is substituted. The result may not be serving to increase the safety or reliability of plants but rather serves to assure the "licensability" of a plant.

On the other hand, there are some observed benefits of the existing licensing process:

1. Incentive - Like preparing for finals or the "big game", nothing seems to inspire participants to complete their homework like a public examination. It is the preparation for the hearing and the struggle to close out open items that brings attention to safety issues that might otherwise go undetected. The hearing itself may be incidental.
2. Attention to Criteria - The General Design Criteria are too often not familiar to the detail designer. An approved hearing draws attention to the design basis and design goals and provides an opportunity for reassessment.

6.3: REVIEW OF SC&A REPORT

The SC&A report covering the licensing process appears to be a well-researched, complete discussion of most licensing issues. [2/] It is, in our opinion, somewhat biased toward the viewpoint of the utility. Put another way, it appears to lack compassion for the concern of the typical intervenor. This is evidenced by the language used to describe the OL process - "a charade" [3/] and characterizing the licensing board as "bending over backwards for the intervenor". [4/]

The report has, however, fairly presented most of the issues, found appropriate fault with many of the current process inadequacies and has fully described a number of potential alternatives or process changes. The following comments are intended to go to the essence of changes that have been proposed.

1. "One step licensing" - A combined proceeding for both CP and OL is a good idea if properly done. This would require a degree of standardization and advance engineering not heretofore performed. Most intervenors would support such a procedure. Intervenors are not attached to the current hearing process, they just want to be assured they will have an opportunity to be heard.

2. Informal meetings - Our experience in such experiments have been positive. An opportunity to exchange ideas has generally narrowed the differences of opinion and has resulted in agreement to settle issues. In addition, an opportunity for honest discussion helps to eliminate misunderstandings and stereotypes.

3. Litigation Alternatives - There are beneficial alternatives to the hearing process. Agreement to an ongoing relationship between local government and the operating plant could have a positive result. This has worked positively in Vermont and has been discussed elsewhere. Provision by the utility of independent design and construction review is a concept that could, in our opinion, supplement or replace the OL hearing.

6.4: CONCLUSIONS

We conclude the following:

1. The existing licensing process has been beneficial and has apparently served (so far) to protect the public.

2. There is a benefit to public participation and the hearing process.
3. A combined CP and OL process would be beneficial if properly conducted.
4. Alternates to the formal process should be pursued. These would include informal public meetings, required independent assessments, and ongoing participation of local government in plant reviews.

SECTION 6

LIST OF REFERENCES

1/ The Future of Conventional Nuclear Power: Nuclear Reactor Regulation (Draft), by Cohen, Lobbin, & Thompson, SC&A, Inc., November 19, 1982.

2/ Ibid 1.

3/ Ibid 1, page 42.

4/ Ibid 1, page 28.

APPENDIX A

CONTRACT WORK STATEMENT

Contract No.:333-2690.0
Exhibit A

STATEMENT OF WORK

The purpose of this contract is to provide OTA with a report in preparation for its workshop on nuclear technology and regulation. This report shall present the perspectives of knowledgeable critics of nuclear power. It is intended to be based on information readily available to the contractor.

The contractor shall identify specific concerns over the safety and reliability of the present generation of reactors. This shall include evaluations of the seriousness of the problems, the difficulty of resolving them, and the degree to which critics would agree that the problems have been resolved.

The contractor shall also evaluate the general approach to the way in which nuclear reactors are designed, built, operated and regulated. This effort will address the question of whether the level of effort is high enough and the quality control stringent enough to ensure that the risks of serious accidents are very low.

The contractor shall also consider the efficacy of alternative proposals that will be discussed at the workshop. For instance, reactors other than the present generation of light water reactors might provide increased safety margins. The contractor shall evaluate the concepts discussed in other OTA contractor reports to determine how useful they might be in resolving the nuclear impasse. Insofar as possible, any new problems that might arise because of a shift to these other technologies shall be noted. The alternative technologies should include reoptimized LWR designs, the ASEA-Atom reactor designed to minimize risks, the High Temperature Gas

Reactor and the CANDU reactor. In addition, the issues of safety, reliability and acceptability as related to size of reactors shall be discussed, though this OTA contractor report may not be finished before December 1.

The contractor shall also discuss the licensing process and various proposals to reform it. This discussion shall focus on the degree to which licensing assures safety, and how the reform proposals might affect that judgement.

Deliverables

A preliminary report shall be delivered by December 2, 1982. The contractor shall present his conclusions at the workshop, to be held December 8 and 9. A final report shall be prepared incorporating any relevent information developed at the workshop. This report shall be due by December 31.

APPENDIX B

SEP DESIGN BASIS EVENTS*

SEP TOPIC LIST OF SAFETY AND ENVIRONMENTAL ISSUES**

* SOURCE: NUREG-0485, Vol. 2, No. 1, August 31, 1979, Page 1-1.

** SOURCE: NUREG-0485, Vol. 2, No. 1, August 31, 1979, Pages 1-2, 1-3, 1-4.

SEP DESIGN BASIS EVENTS

ACCIDENT AND TRANSIENT GROUPS:

GROUP I (PWR):

DECREASE IN FEEDWATER TEMPERATURE (TOPIC XV-1)

INCREASE IN FEEDWATER FLOW (TOPIC XV-1)

INCREASE IN STEAM FLOW (TOPIC XV-1)

INADVERTENT OPENING OF STEAM GENERATOR RELIEF/SAFETY VALVE (TOPIC XV-1)

STARTUP OF INACTIVE LOOP (TOPIC XV-8)

SYSTEM MALFUNCTION CAUSING BORON DILUTION (TOPIC XV-10)

GROUP I (BWR):

DECREASE IN FEEDWATER TEMPERATURE (TOPIC XV-1)

INCREASE IN FEEDWATER FLOW (TOPIC XV-1)

INCREASE IN STEAM FLOW (TOPIC XV-1)

STARTUP OF INACTIVE LOOP (TOPIC XV-1)

INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVE (TOPIC XV-3)

GROUP II:

LOSS OF EXTERNAL LOAD (TOPIC XV-3)

TURBINE TRIP (TOPIC XV-3)

LOSS OF CONDENSER VACUUM (TOPIC XV-3)

STEAM PRESSURE REGULATOR FAILURE (TOPIC XV-3)

LOSS OF FEEDWATER FLOW (TOPIC XV-6)

FEEDWATER SYSTEM PIPE-BREAK (TOPIC XV-6)

GROUP III:

STEAM LINE BREAK INSIDE CONTAINMENT (TOPIC XV-2)

STEAM LINE BREAK OUTSIDE CONTAINMENT (TOPIC XV-2)

GROUP IV:

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B. Population Distribution

* C. Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities

II-2 Meteorology

* A. Severe Weather Phenomena

B. Onsite Meteorological Measurements Program

C. Atmospheric Transport and Diffusion Characteristics for Accidents Analysis

D. Availability of Meteorological Data in the Control Room

II-3 Hydrology

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* B. Flooding Potential and Protection Requirements

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C. Safety-related Water Supply (Ultimate Heat Sink [UHS])

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* B. Proximity of Capable Tectonic Structures in Plant Vicinity

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III-3 Hydrodynamic Loads

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A. Loose Parts Monitoring and Core Barrel Vibration Monitoring

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C. Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves (2)

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IV REACTOR

IV-1 Thermal Hydraulic Design and Performance

A. Operation with Less Than All Loops in Service

IV-2 Reactivity Control Systems Including Functional Design and Protection Against Single Failures

IV-3 BWR Jet Pumps Operating Indications

*Reviewed as part of Design Basis Event evaluation

(1) To be performed independent of SEP as a Division of Operating Reactors generic item.
(2) Not applicable to SEP facilities
(3) To be performed independent of SEP as NRR Category A technical activity.
(4) Completed

◊Consideration being given to deleting topic from SEP review

SEP TOPIC LIST OF SAFETY AND ENVIRONMENTAL ISSUES

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* V-3 Overpressurization Protection (1,3)

V-4 Piping and Safe End Integrity

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* V 7 Reactor Coolant Pump Overspeed

* V 8 Steam Generator (SG) Integrity (1,3)

V-9 Reactor Core Isolation Cooling System (BWR)

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A. RHR Heat Exchanger Tube Failures

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V-11 High Pressure/Low Pressure Interface

A. Requirements for Isolation of High and Low Pressure Systems

B. RHR Interlock Requirements

V-12 Reactor Water Cleanup System (BWR)

A. Water Purity of Boiling Water Reactor Primary Coolant

* V-13 Water Hammer (1,3)

VI ENGINEERED SAFETY FEATURES

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VI 2 Containment Functional Design

A. Pressure Suppression Type BWR Containments (3)

* B. Subcompartment Analysis

* C. Not applicable to the 11 facilities.

D. Mass and Energy Release for Possible Pipe Break Inside Containment

VI-3 Containment Pressure and Heat Removal Capability

VI 4 Containment Isolation System

VI 5 Combustible Gas Control

VI 6 Containment Leak Testing

VI 7 Emergency Core Cooling System

A. Emergency Core Cooling System Performance

1. ECCS Re-evaluation to Account for Increased Vessel Head Temperature (4)

2. Upper Plenum Injection (4)

3. ECCS Actuation System

4. Core Spray Nozzle Effectiveness

B. ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)

C. ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves

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2. Failure Mode Analysis - ECCS

3. The Effect of PWR Loop Isolation Valve Closure During a LOCA on ECCS Performance

D. Long Term Cooling - Passive Failures (e.g., Flooding of Redundant Components) (4)

E. ECCS Sump Design and Test for Recirculation Mode Effectiveness

F. Accumulator Isolation Valves Power and Control System Design

VI 8 Control Room Habitability

VI 9 Main Steam Isolation

A. Main Steam Line Isolation Seal System - BWR

VI-10 Selected Engineered Safety Features (ESF) Aspects

A. Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing

B. Shared Engineered Safety Features, On-site Emergency Power and Service Systems for Multiple Unit Facilities

VII INSTRUMENTATION AND CONTROLS

VII-1 Reactor Trip Systems (IEEE 279)

A. Isolation of Reactor Protection System from Non Safety Systems, Including Qualifications of Isolation Devices

B. Trip Uncertainty and Setpoint Analysis Review of Operating Data Base (4)

VII 2 Engineered Safety Features (ESF) System Control Logic and Design

VII 3 Systems Required for Safe Shutdown

VII 4 Effects of Failure in Non Safety Related Systems on Selected Engineered Safety Features

VII 5 Instruments for Monitoring Radiation and Process Variables During Accidents (3)

VII 6 Frequency Decay (1,3)

VII 7 Acceptability of Swing Bus Design on RWR 4 Plants

*Reviewed as part of Design Basis Event evaluation

(1) To be performed independent of SEP as a Division of Operating Reactors generic item
(2) Not applicable to SEP facilities
(3) To be performed independent of SEP as NRR Category A technical activity
(4) Completed

SEP TOPIC LIST OF SAFETY AND ENVIRONMENTAL ISSUES

VIII ELECTRIC POWER

VIII 1 Offsite Power Systems

A. Potential Equipment Failures Associated with a Degraded Grid Voltage

VIII 2 Onsite Emergency Power Systems - Diesel Generator

VIII 3 Emergency DC Power Systems

A. Station Battery Capacity Test Requirements

B. DC Power System Bus Voltage Monitoring and Annunciation

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IX 2 Overhead Handling Systems - Cranes

IX 3 Station Service and Cooling Water Systems

IX 4 Boron Addition System

IX 5 Ventilation Systems

IX 6 Fire Protection (1)

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XI RADIOACTIVE WASTE MANAGEMENT

XI 1 Appendix I (1)

XI 2 Radiological (Effluent and Process) Monitoring Systems

XIII OPERATIONS

XIII 1 Conduct of Operations

XIII 2 Safeguards/Industrial Security (1)

XV ACCIDENTS AND TRANSIENTS

* XV 1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve

* XV 2 Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

* XV 3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steamisolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)

* XV 4 Loss of Non Emergency AC Power to the Station Auxiliaries

* XV 5 Loss of Normal Feedwater Flow

* XV 6 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

* XV 7 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

* XV 8 Control Rod Misoperation (System Malfunction or Operator Error)

* XV 9 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

* XV 10 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)

* XV 11 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (1)

* XV 12 Spectrum of Rod Ejection Accidents (PWR)

* XV 13 Spectrum of Rod Drop Accidents (BWR)

* XV 14 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

* XV 15 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

* XV 16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

* XV 17 Radiological Consequences of Steam Generator Tube Failure (PWR)

* XV 18 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

* XV 19 Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

* XV 20 Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)

* XV 21 Spent Fuel Cask Drop Accidents (3)

* XV 22 Anticipated Transients Without Scram (3)

* XV 23 Multiple Tube Failures in Steam Generators

XVI TECHNICAL SPECIFICATIONS (SEP)

XVII OPERATIONAL QA PROGRAM (SEP) (4)

*Reviewed as part of Design Basis Event evaluation.

(1) To be performed independent of SEP as a Division of Operating Reactors generic item
(2) Not applicable to SEP facilities
(3) To be performed independent of SEP as NRR Category A technical activity
(4) Completed

APPENDIX C

STATEMENT OF RICHARD B. HUBBARD

BEFORE

HOUSE SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT

(November 19, 1981)

STATEMENT OF RICHARD B. HUBBARD

BEFORE

HOUSE SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT

(November 19, 1981)

A linchpin in the NRC's "defense-in-depth" approach to nuclear safety is its emphasis upon Quality Assurance (QA). The NRC in its 1980 Annual Report emphasized that ". . .

> the application of disciplined engineering practices and thorough management and programmatic controls to the design, fabrication, construction, and operation of nuclear power plants is essential to the protection of public health and safety and the environment. QA provides this necessary discipline and control. (emphasis added).

In simplicity the QA program requires that you plan what you do, do what you plan, and document what you did.[1/]

QA requirements for nuclear power plant design and construction have expanded greatly since the mid-1960's when the Diablo Canyon project was initiated. Earlier methods of ensuring quality were largely informal, guided by a modest set of codes and standards. Starting in 1970, however, a new approach was developed requiring adherence to a number of specific, detailed NRC rules and industry standards.[2/] These rules and standards had

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1/ My qualifications and experience as a QA Manager and a consultant are summarized in Attachment A, which is appended to this statement.

2/ There is no question that the NRC's QA criteria had a significant impact on utility work practices. For example, the Vice President of Engineering for Long Island Lighting Company stated:

> . . . on fossil facilities construction engineers had considerable latitude in resolving installation problems. This latitude is not allowed

(cont'd. on page 2)

a basic purpose: to ensure that the design, construction and operation of nuclear power facilities met high quality standards, thus providing the assurance that the plants would generate power safely.

In the past few years, however, there have occurred a series of QA program breakdowns which raise serious generic questions about the ability of the NRC licensing review and inspection process to provide assurance that these QA rules have been effectively implemented. These QA breakdowns include:

-- South Texas Project -- multiple design flaws as set forth in the recent Quadrex report following earlier disclosure of widespread construction flaws;

-- Stone & Webster Seismic Models -- incorrect seismic analysis resulting in a 5 plant shutdown followed by a series of NRC bulletins.

The Diablo Canyon QA errors are the most recent example.

From October 14-23, 1981, I accompanied the NRC personnel from Region V in their special inspection to evaluate the Diablo Canyon QA program and other management control systems in effect at PG&E during the period from 1970 to the present. The inspection

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2/ (cont'd.)

> for a nuclear plant. According to Appendix B of 10 CFR 50, 'design changes including field changes must be approved by the organization which performed the original design.' As a result, the construction organization at Shoreham has generated over 33,000 design changes reports that required disposition and approval by the Engineering organization.
>
> Direct testimony of Dr. Matthew C. Cordaro, Vice President of Engineering for LILCO, NY PSC Case # 27563, Nov. 2, 1981, p. 160.

revealed widespread deficiencies in the Diablo Canyon QA program and its implementation. These deficiencies, which went undetected by PG&E and the NRC for years, involve a variety of items important to safe operation of Diablo Canyon. The purpose of this statement is to briefly summarize my findings -- findings both with regard to Diablo Canyon and to QA program implementation and NRC review in general.

In the six weeks since the Diablo Canyon fuel loading and low power testing license was authorized on September 21, fourteen (14) separate errors in seismic design have been discovered at Diablo Canyon.[3/] These significant errors have primarily involved breakdowns in the development, distribution, and use of design data by PG&E and its engineering services subcontractors. The significance of these serious QA errors cannot be disputed. Indeed when only the first two of the errors now known were revealed, Dr. Harold Denton of the NRC Staff stated that the low power license would not have been issued if these errors had been known to the Staff before the NRC issued the license.[4/] The 14 errors involved:

-- A Unit 2 diagram of the containment annulus area was erroneously used in the seismic analysis of Unit 1.

-- Latest design data were not provided to the engineer responsible for the seismic design of conduit and cable tray supports.

-- Weights and weight distribution for piping and equipment in the containment area were substantially underestimated.

3/ A list of the 14 separate errors is included herein as Attachment B.

4/ October 9 NRC Staff/PG&E meeting transcript, p. 117.

EDITOR'S NOTE

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4/ October 9 NRC Staff/PG&E meeting transcript, p. 117.

-- Containment spray system pipe support design was deficient in part due to an incorrect thermal analyses and an incorrect assumption that the pipes contained water.

-- Discrepancies were discovered between the "as-built" and "as-designed" configuration of supports for electrical conduits.

-- Repeated instances of misapplication and incorrect use of Hosgri earthquake design data were disclosed. Errors in addition to bad judgment included selecting the wrong number from the earthquake response curve and using the earthquake spectra for the wrong location in the building.

My experience suggests that, given the succession of errors already disclosed, further investigations will almost certainly reveal more errors.

The eighteen QA requirements of Appendix B to 10 C.F.R. Part 50, which were enacted by the NRC in 1970, are specifically designed to detect such errors and thus ensure that nuclear plants are designed and constructed in accordance with all requisite requirements. PG&E did not detect these errors because it did not comply with Appendix B. While the root cause of many of PG&E's errors has not yet been determined, it is clear that a significant contributor to the many instances of "informality" in the development, distribution, and use of safety data discovered by the NRC was PG&E's failure to develop detailed QA procedures until the 1976-1977 period.[5/]

5/ NRC Inspection Report 50-275/81-29, see pp. 3-6.

Based on my experience in developing the QA program for nuclear plant equipment at GE, I believe that by 1972 the requirement for a well defined QA program including detailed implementing procedures was generally recognized. Such a program with procedures which demonstrated how the program was to be implemented was not developed by PG&E for Diablo Canyon until nearly five years later. As a result, the required QA activities for design and construction either did not occur, or did not occur in a consistent and prescribed manner. Examples of PG&E's QA violations included:

-- PG&E's major service subcontractor, John Blume and Associates, was not contracturally obligated to a QA program until 1978, eight years after Appendix B was adopted and twelve years after Blume's first engineering services on Diablo Canyon. Criterion 2 requires that a QA program be established "at the earliest practicable time."

-- Unverified and uncontrolled design data were provided to Blume in violation of Criteria 3, 4, and 5 regarding design control, procurement document control, and documentation.

-- Document distribution measures did not assure, as required by Criterion 6, that documents intended for use internal and external to PG&E "are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed."

-- Qualification and evaluation of service contractors did not occur as required by Criterion 7.

-- Audits of suppliers, such as Blume, were not carried out in a timely fashion to verify compliance with the QA program and to determine the effectiveness of the program. This violated Criterion 18.

Full compliance with the NRC QA requirements is of fundamental importance to safety. There can be no doubt as to whether PG&E and its subcontractors have developed and implemented a QA program in full compliance with the regulatory criteria -- they have not. Thus, there is substantial uncertainty surrounding the actual quality level achieved in the design and construction of Diablo Canyon.

In recognition of the PG&E QA shortcomings, it is imperative that an independent audit be initiated. The audit should include two vital elements: (1) the audit should provide for a physical reinspection of plant equipment as well as a design reverification of Diablo Canyon safety systems; and (2) the audits, and any plant modifications, should be completed prior to plant operation. The two key elements are set forth in more detail in the following.

First, the audits must address construction QA. The NRC has assumed that "the weaknesses in the [QA] program are focused in the service contract area and that it is not a universal breakdown throughout the company."[6/] There is insufficient data to support this conclusion. Rather, repeated QA program breakdowns have been found in all areas subject to the NRC's narrow reinspection program. In addition, there is evidence that construction QA may have experienced the same basic problems as design QA:

-- Construction QA was covered by the same QA manual as design QA, which has been shown to have been inadequately planned and implemented.

6/ November 9 NRC Commission meeting transcript, p. 22.

-- In 1980, PG&E reported that its inspections pursuant to I&E Bull. 79-14 revealed that 26 percent of as-built piping required reanalysis. See Attachment C hereto, p. 2. Dr. Cloud found similar discrepancies between "as-built" and designed configuration of supports for electrical conduits. This indicates a clear potential problem in construction QA or, at least, in the interface between design and construction QA.

-- There have been numerous past instances of construction defects at Diablo Canyon and allegations of construction QA problems. Examples included inadequate training of welders and radiographers, widespread defects in welding of pipe supports and pipe whip restraints, and over 10,000 defective anchor bolt installations.

Since there can be no confidence in the quality of the "paper" documenting the Diablo Canyon design and its construction implementation, it is therefore imperative that the actual quality of installed safety-related structures, systems, and components be established. The means to do so would be through a physical reinspection of a few critical electrical and mechanical systems. There is precedent for such a reinspection. Currently, Long Island Lighting Company (LILCO) is proposing such a reappraisal for the Shoreham plant in response to concerns expressed by the Suffolk County legislature.[7/] Such a physical reinspection, when combined with the audit areas proposed by the NRC, would appear to provide a meaningful measure of the quality of the plant as well as the quality of the paper.

7/ Guidelines for an independent physical reinspection and design verification audit program for Diablo Canyon were attached to Governor Brown's October 30 letter. The proposed audit for Diablo Canyon is similar to that for Shoreham, but reduced in scope by not including a pressure vessel inspection.

Second, Diablo Canyon should not be made radioactive prior to completion of the independent audits. This means that the plant should not operate until any required plant modifications are completed. Enclosed herewith as Attachment D is a July 29, 1979, PG&E QA audit report of the PG&E/Blume contract covering the post-1977 period. The report documents a continuing and significant breakdown of QA at URS/Blume and at PG&E during the period following the Hosgri reanalysis. The following are excerpts from the PG&E report:

> The auditors found that in 12 instances URS/Blume was not implementing contractual and Quality Assurance Manual requirements. Due to the significant nature of the audit findings, it was concluded that URS/Blume had not been responsive to its commitment for a quality program. (emphasis supplied).

The deficiencies which were identified covered a wide range of Blume's activities:

> The deficiencies occurred in the following areas: design review, internal audits, annual reports to management, organization chart, audits of Document Control Center, computer program verification, and interface procedures.

The deficiencies also directly involved PG&E's internal QA procedures, both with respect to the Blume contract and other service contracts.

> There is no clearly defined interface between URS/Blume and the Engineering Department as required by ANSI N45.2.11 (§§ 2.2 & 5.1). Additionally, the Engineering Department manual does not presently require that design interfaces be identified and controlled between PGandE, URS/Blume, and other architect-engineers as required by the Standards. (emphasis supplied).

Thus, the audit found that both the Blume and PG&E QA program were deficient in significant respects.

> The deficiencies identified are not minor and indicate that a problem exists in the implementation of URS/Blume's quality assurance program. In addition to the URS/Blume deficiencies, problems were identified within the interfacing activities between PGandE and its consultants. (emphasis supplied).

The foregoing demonstrates that the serious QA problems within PG&E and with Blume were not confined to the 1977 Hosgri reanalysis period, but continued at least until mid-1979. Further, in mid-1979, PG&E still was failing to interface properly with its service contractors.

Therefore, I conclude that a design reverification of all safety systems should be completed before Diablo Canyon operation is permitted at any power level. Further, I recommend that the scope of the necessary audits should include construction QA. Finally, I believe that the results of the proposed independent physical reinspection, when combined with the design reverification will be a significant step in determining the quality of the Diablo Canyon plant and in providing the public with a credible technical analysis in which the public can have confidence.

ATTACHMENT A

EXPERIENCE AND QUALIFICATIONS OF RICHARD B. HUBBARD

Richard B. Hubbard is a Professional Quality Engineer licensed by the State of California (license number QU 805), a technical consultant, and a founder (in 1976) and vice president of MHB Technical Associates, a corporation engaged in the business of technical consulting on energy and environmental issues, having its principal office at 1723 Hamilton Avenue, San Jose, California, 95125. Hubbard holds a B.S. in Electrical Engineering from the University of Arizona (1960) and an M.B.A. from the University of Santa Clara (1969). He has seventeen years experience in the design, manufacture, construction, and operation of nuclear power generation facilities including eleven years experience in responsible managerial positions in the Nuclear Instrumentation Department (1965-1971), Atomic Power Equipment Department (1971-1975), and Nuclear Energy Control and Instrumentation Department (1975-1976) of the General Electric Company (GE). For the past five years, Hubbard, along with his co-founders of MHB Technical Associates, have conducted numerous studies pertaining to the safety, quality, reliability, and economic aspects of nuclear power facilities.

From November, 1971 to February, 1976, Hubbard was Manager of Quality Assurance for the manufacturing operations at the San Jose, California, headquarters of GE's Nuclear Energy Division. He was responsible for the development and implementation of quality plans, programs, methods, and equipment to assure that equipment for nuclear plants manufactured and procured by General Electric met quality requirements as defined in NRC regulation 10 C.F.R. 50, Appendix B; ASME Boiler and Pressure Vessel Code; customer contracts; and GE Corporate policies and procedures. While employed by General Electric, Hubbard was responsible for developing a quality system which received NRC certification in 1975. The system was also successfully surveyed for ASME "N" and "NPT" symbol authorizations in 1975. He was also responsible for the quality assurance program and its implementation at GE's spare and renewal parts warehouse.

Hubbard is a member of the IEEE Nuclear Power Engineering standards subcommittee responsible for the preparation and revision of Quality Assurance standards for safety-related aspects of nuclear power facilities. He is currently a member of the IEEE Committee which is preparing a standard which addresses the re-

quirements, including the quality assurance program, for the selection and utilization of replacement parts for Class IE equipment during the construction and operation phase.

Hubbard has testified on safety-related aspects of nuclear power facilities' quality assurance programs as an expert witness before the Nuclear Regulatory Commission Atomic Safety and Licensing Boards; before (and at the request of) the NRC's Advisory Committee on Reactor Safeguards; before the Joint Committee on Atomic Energy of the United States Congress; and before various federal and state legislative and administrative bodies.

ATTACHMENT B

QUALITY ASSURANCE ERRORS REVEALED
SINCE SEPTEMBER 21, 1981

In the six weeks since September 21, 1981, 14 serious errors in seismic design have been discovered at Diablo Canyon. These errors have primarily involved problems in the development, distribution, and use of design data by PG&E and its engineering services subcontractors. These errors were discussed at Commission and Staff meetings in Washington, D.C. and Bethesda, Maryland and at a series of meetings during October, 1981, with PG&E in San Francisco, California. As a result of these discussions and investigations, it is now clear that each error involved a failure of PG&E to implement properly a number of the 18 quality assurance criteria of 10 C.F.R. Part 50, Appendix B. The errors are described below:

a. Error 1 - Opposite Hand Design

On September 28, 1981, PG&E reported that a diagram error had been found in a portion of the seismic qualification of the Diablo Canyon Unit 1 Nuclear Power Plant (DCPP-1). This error resulted in an incorrect application of the seismic floor response spectra in the crane wall-containment shell annulus of the Unit 1 Containment Building. The error was that the diagram used to locate Vertical Seismic Floor Response (VSFR) spectra for the Unit 1 containment annulus was erroneous. The diagram was applicable to Unit 2 but was identified as being that of Unit 1. Since the Units are opposite hand, this resulted in an incorrect orientation of VSFR spectra for Unit 1 component and system design. The origin of the error was in the PG&E transmittal, to a subcontractor (John A. Blume and Associates), of an unverified, handwritten sketch of the Unit 2 opposite hand geometry in place of the Unit 1 geometry. 1/ (Also see Error 3).

b. Error 2 - Document Distribution

At the October 9 meeting between the NRC Staff and PG&E, PG&E disclosed that the Seismic Category I electrical cable trays and conduit supports had been qualified to design response spectra which had been superseded. The error was caused by PG&E's failure to distribute the latest revised spectra to the responsible engineer. 2/

1/ LER 81-002/01T-0, October 12, 1981.

2/ October 9 meeting transcript, pp. 105-07.

c. Error 3 - Incorrect Weights

On October 22, 1981 inspectors from the NRC's Office of Inspection and Enforcement determined that, in addition to the improper application of the diagram as reported to the NRC by PG&E on September 28, 1981, the weights listed on the diagram and used as an imput to John A. Blume and Associates for their development of response spectra, could not be verified as being accurate. PG&E representatives recalculated the weights, using current as-built drawings, and determined the new weights to be different.[3/]

PG&E concluded that the "substantial" weight variations resulted from three principal causes:

(i) The large bore piping equipment weights were not associated with the correct frames because the Unit 1 piping orientation was used in conjunction with the Unit 2 frame orientation.

(ii) PG&E's current calculations include additional contributors to the total weight: e.g., conduit and cable trays and steel grating, which were considered to be insignificant in the 1977 analysis.

(iii) A more detailed calculation of large bore piping weights, piping support weights and equipment weights.

d. Error 4 - Containment Spray System Pipe Supports

On September 18, 1981, the NRC's resident inspector was notified by telephone of a deficiency reportable under 10 CFR 50.55(e). [4/] The report from PG&E addressed deficiencies in the design of the containment spray system pipe supports located within containment.

The following four deficiencies were identified:

(i) An incorrect thermal analysis was used for hanger loads.

3/ PNO-V-81-59, Preliminary NRC Notification of Event or Unusual Occurrence, October 26, 1981.

4/ Letter from Crane to Engelken, October 19, 1981. Inexplicably, this error was not brought to the NRC's Commissioners' attention September 21, 1981.

(ii) This analysis was performed with one snubber modeled as a rigid member.

(iii) The variable spring settings for the pipe supports were improperly set, based on a deadload analysis which assumed, incorrectly, that the pipes contained water.

(iv) In designing a pipe anchor, the loads from only one side were used.

The root cause of the preceding series of errors has not yet been explained by PG&E.

e. Error 5 - Wrong Spectra

During the period of October 14 through 16, 1981, representatives of the NRC staff and their consultants from Brookhaven National Laboratories met with the PG&E staff in San Francisco. During the meeting, piping problem (PG&E #6-11) was reviewed. PG&E initially asserted that this problem did not require reevaluation as a result of the opposite hand error. However, it was subsequently determined that the original PG&E calculation used erroneous spectra input and hence required reanalysis with the appropriate spectra. 5/ The cause of the error has not yet been identified by PG&E.

f. Errors 6 to 10 - Additional Design Errors

At the November 3 meeting between PG&E and the NRC, PG&E disclosed that during its internal review undertaken as a result of the diagram frame orientation error, it has identified five additional design errors requiring plant modifications from causes not related to the diagram error. These design deficiencies are:

(i) In a single case, parallel piping lines which were qualified and designed from a single analysis actually require two analyses to properly model both configurations.

5/ NRC Meeting Summary for October 14-16, 1981, Discussions and Preliminary Audit of Seismic Analysis for Equipment and Components in Diablo Canyon, Unit 1 Containment Annulus, p. 4.

(ii) In two cases, a small bore piping snubber required by analysis was not designed.

(iii) Two supports contained gaps insufficient for thermal movement.

(iv) A center support was not rigid in the restrained direction.

(v) A single support on a nonsafety-related pipe had not been qualified to Hosgri loads as required to prevent interaction with an adjacent safety-related pipe. 6/

The cause of four of the five design errors has not yet been determined by PG&E. The first error resulted from an error in judgment. The engineer looked at the two parallel-component cooling lines which run around the annulus, picked what he thought was the worst case, analyzed that worst case to determine pipe stressing and support loadings, and based on that the supports were designed for both lines. PG&E subsequently determined that the engineer had not used a conservative approach in that the engineer did not pick the worst case. Rather, the line he did not use as a model for the analysis had a longer riser section, and so it was necessary to perform a separate model for that other line. 7/

g. Error 11 - Incorrect Vertical Spectra

A discrepancy in the spectra that were applied to the regenerative heat exchanger was discovered during Dr. Cloud's review. The error was that the engineer responsible for qualifying this equipment used two-thirds of the horizontal Hosgri acceleration filtered for the tau effect. In fact he was supposed to have used two-thirds of the unfiltered spectra. The underlying cause of the error has not yet been determined by PG&E. 8/

6/ November 3 meeting transcript, pp. 138-140.

7/ November 3 meeting transcript, p. 142.

8/ November 3 meeting transcript, p. 201.

h. Error 12 - Misapplications of Hosgri Spectra

Error 12 involved misapplication of the Hosgri spectra. Electrical raceway and conduit supports are unistrut type supports, are all Class 1 equipment, and are all laterally braced. The PG&E seismic analysis is based on an enveloping procedure using static analysis. In this analysis, which contains a large number of configurations, the largest weight that a particular configuration is considered to be able to have applied to it is determined, and the highest acceleration the support can experience owing to its location in the building, is also determined. Then, with those two inputs, the first mode frequency of the supports is calculated, and the corresponding acceleration level is taken from the response spectra and the stress analysis is conducted. The misapplication errors were basically of two kinds. First, the analyst selected the wrong number off the response spectra curve; and second, in some cases the engineer apparently used one of the Hosgri spectra from a different location in the building. As before, the cause of the errors has not yet been determined by PG&E. 9/

i. Error 13 - Further Spectra Misapplication

For the heating and ventilating system components, Dr. Cloud reviewed the seismic input for the fans and dampers. He found one instance where the Hosgri spectra were misapplied. Once again, the manner in which this analysis was conducted is very similar to that for the conduit supports (Error 12). The engineer confirmed that the equipment was rigid, and then went to the zero portion of the response spectra curve and selected the wrong value for the acceleration level. In this case, PG&E believes that the engineer used a spectra from a different location of the building. The cause of the error has not been determined by PG&E. 10/

j. Error 14 - Differences Between "As-Built" and Design Conditions

Several instances of difference were discoverd between the "as-built" and "as-designed" configuration of supports for electrical conduits. The discovery of these discrepancies once again demonstrates the necessity for field verification of plant conditions as compared to the models used for seismic analyses. The cause of the errors have not yet been determined by PG&E. 11/

9/ November 3 meeting transcript, pp. 204-205.

10/ November 3 meeting transcript, p. 206.

11/ Progress Report No. 1, Seismic Re-Verification Program by C. L. Cloud, November 11, 1981, pp. 5 and 8.

ATTACHMENT C

PACIFIC GAS AND ELECTRIC COMPANY

LAW DEPARTMENT - 77 BEALE STREET, 31ST FLOOR • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-421

April 17, 1980

Mr. R. H. Engelken, Director
Office of Inspection and Enforcement
Region V
U. S. Nuclear Regulatory Commission
1990 N. California Boulevard
Walnut Creek Plaza, Suite 202
Walnut Creek, California 94596

Re: Docket No. 50-275
Docket No. 50-323
Diablo Canyon Units 1 and 2

Reference: U. S. NRC Bulletin 79-14
dated July 2, 1979

Dear Mr. Engelken:

This letter is written in response to the referenced IE Bulletin and supplements which require a plant inspection to verify that actual configurations of safety-related piping agree with the models used to seismically analyze them.

Our letter of October 17, 1979 submitted our response to Item 1 of the bulletin. This letter responds to Items 2, 3, and 4 for Diablo Canyon Unit 1 and is our final response for Diablo Canyon Unit 1. Review of Diablo Canyon Unit 2 and the attendant report will follow at a later date.

Items 2 and 3 call for a field inspection of as-built piping configurations, a comparison with the corresponding seismic analysis models, and a description of the results of the review. Item 2 requests that this be done for normally accessible systems, and Item 3 requests the same action for inaccessible systems. Since Diablo Canyon is not yet an operating plant all systems are accessible and no distinction was made between Items 2 and 3. Systems represented by a total of 75 piping isometrics covering all seismic Class 1 piping 2-1/2 inches and larger and 56 isometrics covering all seismically analyzed 2-inch and smaller piping were inspected by field personnel. They were instructed to yellow over items in agreement with the drawings and to mark in red all differing information. This inspection

was completed on November 9, 1979.

Following the inspection, the isometrics were compared point by point with 192 large diameter (greater than 2-1/2 inch) piping seismic analyses and 30 small diameter piping seismic analyses. Some of the piping, such as the reactor coolant loops, was analysed by Westinghouse, which performed the 79-14 review for these systems. For each analysis, a package was prepared comparing support locations and directions, pipe geometry, valve data, materi properties and other information. A list of discrepancies was compi and each discrepancy was reviewed by a team of engineers to assess i impact and to determine an appropriate course of action. This revie was completed February 15, 1980.

The following types of discrepancies are typical of those found, in order of frequency of occurrence: valve weights not correct; weights of valve flanges not modelled; center of gravity of valve operator not adequately considered; support location differenc of greater than one pipe diameter; supports missing or extra; presen of high density lead foam or grout in penetrations; differences in p geometry; invalid assumptions in modeling of analysis endpoints; dif ferences in insulation thickness and pipe diameter.

It was decided that 49 of the 192 large diameter analyses and 8 of the 30 small diameter analyses had differences significant enough that the results were not obviously conservative and that the should be reanalysed. This amounted to approximately a 26 percent reanalysis rate. In addition, there were 10 large diameter and 4 small diameter analyses for which differences were resolved by a field hardware change. For the Westinghouse-analysed lines, althoug some minor disagreements between the analysed and the as-built condi tion were found, Westinghouse determined that none of these disagree ments were serious enough to warrant reanalysis.

Item 4 requires an evaluation of the effect of the discrepancies on system operability, a schedule for completion of reanalysis, and a description of procedures in effect to assure timely incorporation of as-built changes. The reanalysis was completed on April 11, 1980. The results of the reanalysis were as follows: Two of the analyses showed an overstress in the pipe due to the discrepancies found, and additional supports were added to correct the problem. Twelve of the analyses showed no significant increase in load, deflection or stress. The remaining 43 analyses had no proble with respect to pipe stress but had increases in support reactions. At this time, all supports which have increased design loads are bei rechecked to determine whether structural modifications are needed. At present about 20 supports have been identified as needing rework. Reevaluation of supports will be completed approximately May 1, 1980. Field modifications will be finished approximately June 1, 1980.

Following completion of the review and rework, the inform tion obtained in the field inspection will be incorporated into the design review piping isometrics. Pipe support detail drawings will also be updated to reflect as-built conditions.

Engineering procedures in force at this time require that as-built drawings of safety-related piping isometrics and pipe supports be made and be submitted to the engineering staff for review. As-built drawings are forwarded to the support designer, who checks to ensure that variations from the original design do not impair th intended function of the system or its structural adequacy. The designer then updates his records to reflect the as-built condition Procedures also require that changes in pipe routing, changes in equipment or changes in support location in safety-related systems are not made unless prior engineering approval is obtained. Such approval is given only after applicable analyses are consulted. We feel that these procedures for as-built documentation assure that i the future there will be strict conformity between the physical con figuration of safety-related piping systems and the related design documents.

Very truly yours,

Richard F. Locke

cc: Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Service List

PACIFIC GAS AND ELECTRIC COMPANY

PG&E — 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

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June 29, 1979

Mr. R. F. Runge
Quality Assurance Manager
URS/John A. Blume & Associates, Engineers
130 Jessie Street
San Francisco, CA 94105

Dear Bob:

I performed an audit at your San Francisco offices during the period of April 23 through May 24, 1979. The purpose of this audit was to confirm that URS/Blume is implementing contractual and Quality Assurance requirements. You are aware of the problems and deficiencies which we encountered as well as the length of time and man-hours needed to verify certain aspects of your quality program.

While the audit report has been written, the resolution to several audit findings remains to be verified; in general, I expect them to be resolved with the implementation of Revision 7 to your QA Manual.

Some of the deficiencies in your program may seem minor, but it is PG&E's responsibility to verify operation of your program to our satisfaction. For this reason, I requested the assistance of several engineers in reviewing various technical aspects of your projects in order to verify the accuracy and quality of the work.

In the future, PG&E expects your Quality Assurance Program to be fully functional and all your project personnel familiar with the applicable quality requirements of their projects. A follow-up audit will be conducted later this year to verify the resolution of all outstanding findings.

Sincerely,

M. L. Barham

MLB(3692):rr

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DISTRIBUTION OF AUDIT REPORTS

In accordance with Quality Assurance Department Procedure 11.8, "Audit Reports: Form, Content, and Distribution", audit reports are to be submitted to the Director, Quality Assurance, within 2 weeks (10 workdays) of the conclusion of the audit.

If the above schedule is not met, explain below:

Audit Report No.: 91605

Conclusion Date of Audit: MAY 24, 1979

Issue Date of Audit: JUNE 29, 1979

Reason for Delay: Due to unusual nature of a
based upon Supplier Qual
program implementation — ma
additional days of investigation
were required to establish
the credability of work perf
by the Supplier.

Approved: _____
W. J. Raymond

QUALITY ASSURANCE
11.1 x 8.0

Audit of URS/Blume & Associates, Engineers
San Francisco, California
Audit No. 01605

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June 29, 1979

MR. B. W. SHACKELFORD:

Attached is the report of an audit of URS/Blume performed on April 23 through May 24, 1979. URS/Blume provides consulting services for seismic qualification of structural (and mechanical) supports at Diablo Canyon Power Plant. This work includes design reviews, analyses of support systems, and new design for modifications.

The audit was performed to verify that this consultant was adequately implementing its quality assurance program on the PGandE projects for which it is responsible.

The auditors found that in 12 instances URS/Blume was not implementing contractual and Quality Assurance Manual requirements. Due to the significant nature of the audit findings, it was concluded that URS/Blume had not been responsive to its commitment for a quality program. Immediate action was taken to inform Project Engineering of the findings, and to schedule corrective actions.

After the initial findings, PGandE engineers were asked to assist the audit team to verify the quality of URS/Blume's design work. In general, the design work itself appeared to be satisfactory. Two computer programs which had not been verified have since been verified to comply with regulations.

In addition to the deficiencies identified in the URS/Blume QA Program, deficiencies in PGandE's interfacing activities with consultants were also identified. All findings have been documented on Nonconformance Reports #DCO-79-EM-004, -005 and -006. Corrective actions will be verified by the Quality Assurance Department.

M. A. GILMORE

TCU/HLMarham (3692):rr

cc: DC & HB - Supplier

Attachment

3/9/79

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BPSadler
JOSchuyler
BWShackelford (original letter)
EDWeeks
JDWorthington

PACIFIC GAS AND ELECTRIC COMPANY
QUALITY ASSURANCE DEPARTMENT

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Title: Audit of Supplier's Quality Assurance Program

Audited Organization/ Facility: URS/John A. Blume & Associates, Engineers

Auditors: M. L. Barham (Lead Auditor)
C. L. Eldridge

Dates Performed: April 23-May 24, 1979

1.0 Scope

URS/Blume has been providing engineering consulting services, including design, review, and analysis, for seismic qualification of structural supports. This audit was performed to verify implementation of all aspects of URS/Blume's quality assurance program related to their present work at Diablo Canyon Power Plant.

2.0 Conclusion

On May 24, 1979, a post-audit conference was held at URS/Blume's facilities. Twelve audit findings were presented and discussed. The following people were in attendance:

PGandE	URS/Blume	
M. L. Barham	Betty Anderson	C. Kansler
C. L. Eldridge	B. Chakravartula	H. Klice
M. E. Lee	D. Dilip	D. A. Lang
J. J. Benitou	R. Gallagher	P. K. Lum
	G. Hess	B. Mullin
	K. Honda	R. F. Runge
	K. Jay	R. Yokoyama

The audit findings were discussed and audit finding reports presented to URS/Blume's management. The deficiencies occurred in the following areas: design review, internal audits, annual reports to management, organization chart, audits of Document Control Center, computer program verification, and interface procedures. Details of the audit findings are in Appendix A.

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The deficiences identified are not minor and indicate that a problem exists in the implementation of URS/Blume's quality assurance program. In addition to the URS/Blume deficiencies, problems were identified with the interfacing activities between PGandE and its consultants. The Qual. Assurance Department has issued three Nonconformance Reports to document these audit findings and will follow their progress to resolution.

3.0 Nonconformance Reports (NCRs) Initiated During This Audit

3.1 NCR #DCO-79-EN-004
Responsible Department: Engineering
Description: The URS/Blume QA program for work on Diabl Canyon has not been effectively implemente Audit Findings Nos. 91605-1 through 91605- have been issued to URS/Blume.

3.2 NCR #DCO-79-EN-005
Responsible Department: Engineering
Description: URS/Blume's QA program not effectively implemented and as such is unable to provi assurance that design work performed is adequate.

3.3 NCR #DCO-79-EN-006
Responsible Department: Engineering
Description: No clearly defined interface between URS/B and the Engineering Department as required ANSI N45.2.11 (§ 2.2 & 5.1). Engineering Department manual does not require that design interfaces be identified and contro between PGandE, URS/Blume, and other architect-engineers.

Additional details of these NCRs are in Appendix A.

Performed by: M. L. Barham
M. L. Barham

C. L. Eldridge by T
C. L. Eldridge

Approved by: W. A. Raymond
W. A. Raymond

Attached: Appendix A

APPENDIX A

AUDIT ACTIVITIES

1.0 References

(1) 10CFR50, Appendix B
(2) ANSI N45.2.9-1974
(3) ANSI N45.2.11-1974
(4) ANSI N45.2.23-1978
(5) URS/John A. Blume & Associates, Engineers Quality Assurance Manual, Revision 5, 6/9/78

2.0 Documents Reviewed

(1) Training records
(2) Quality Assurance Manual revision control records
(3) Design calculation packages
(4) G-Line Anchor Design Review Package
(5) Specifications and design criteria
(6) Computer file registers, program logs, and certified program list
(7) Project file registers
(8) Document Distribution Lists
(9) Design Review Reports
(10) Audit Deficiency Reports

3.0 Audit Findings

3.1 Supplier Audit Finding Reports

	Finding	Recommendation
(1)	Independent review of critical aspects of URS/Blume work is required to be performed by a Quality Assurance Engineer prior to transmittal of work to client. Such reviews have not been performed on the bulk of work prepared for PGandE. Documents have been transmitted to PGandE, accepted for construction, and used to perform work at the site.	Obtain written direction from PGandE specifying the actual design review completion dates for each of the projects at URS/Blume Engineers.
(2)	Internal audits have not been performed in accordance with frequency requirements described in URS/Blume's Procedure 6.1.	URS/Blume must conform to its QA requirements for audit scheduling; Blume must perform an audit on each of the projects it is working on for PGandE.

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	Finding	Recommendation
(3)	Requirements for annual program review and report to management have not been met for 1978.	Comply with requirements; pe program review and prepare a report to management on the status of the QA program including achievements as we as problems.
(4)	Organization chart does not accurately represent the present URS/Blume organization.	Revise organization chart to accurately represent present lines of authority and responsibility.
(5)	URS/Blume's Quality Assurance Manager has not performed the required audits of the Document Control Center.	Perform regularly scheduled audits of Document Control C as required by the QA progra
(6)	Several major computer programs used for analysis of PGandE projects do not have the required verification and certification. This information is required to be filed in the user manual kept by the Computer Services Manager, and its availability is of extreme importance.	Complete documentation, as required by QA Manual, for a computer programs. PIPESD- (version 5.2 and 5.3) must ha the procedure and method used to verify.
(7)	Several less critical program routines used on project piping analysis have not been certified or verified. The background data for MODPIP, ENVELPS, and RRIPS (Rapid Reanalysis) are kept by the Project Manager; however, the QA Manual states that the Computer Services Manager is responsible for maintenance of all program documentation.	Complete documentation, as required by QA Manual, for all computer programs used on PGandE projects. Modify Manu to explain how some programs, unique to a project, are maintained and stored with th project files and not with th Computer Services Department.
(8)	Interface procedures have not been written or approved as required by Procedure 4.6.	Write interface procedures to identify interactions between URS/Blume's office, its field forces, and PGandE. Procedur to be approved by QA Manager. Perform audits of field activi ties to verify compliance with the QA program.

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4.0 Additional Information

4.1 PGandE Personnel Accompanying Audit Team

R. E. Bacher	M&NE
J. J. Benitou	EQC
V. M. Chauhan	M&NE
V. J. Ghio	CE
S. D. Krein	EQC
M. E. Lee	CE
G. R. Veerkamp	M&NE

4.2 Additional PGandE Personnel Contacted

D. J. Curtis
J. B. Hoch
C. M. Li
J. J. Kilian
A. G. Walther
E. P. Wollak

4.3 URS/Blume Personnel Contacted

B. Anderson
B. Chakravartula
P. Chokkalingan
R. Gallagher
G. Hess
K. Honda
K. Jay
C. Kensler
H. Klice
D. Lang
P. Lum
B. Mullin
C. Quan
R. Runge
S. Swan
R. Yokoyama

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4.4 Additional Findings by Engineering Department

Several PGandE engineers were informed of the state of URS/Blume's QA program and asked to accompany the audit team. The intent was to determine if the lack of implementation of the QA program had resulted in actual design problems and questionable results or if it resulted mainly in documentation omissions.

The engineers involved in the audit were directed to critically examine selected portions of URS/Blume's work to verify that it mee the requirements of the design criteria.

Mr. M. E. Lee (C.E.) was responsible for reviewing design verificat performed on the Intake Structure analysis. He verified that the computer programs used on this project (TABS and SAP IV) were verif per the QA program. He reviewed the hand calculations and discover that the last three sections (of ten sections) had been prepared by the project manager, D. Lang, and not checked.

These sections dealt primarily with answers to NRC questions on Hos Seismic Analysis which had been prepared under difficult scheduling constraints. The project manager has stated that he will either ha the calculations checked (as the bulk of the work has been) or plac statement in the file that the design review verified the calculati This item will be followed with SAFR #91605-10.

The PGandE Engineer checked the design review package, assumptions, dimensions of structures, computer output file, and the summary of Final Revised Design Analysis with incorporated changes from URS/Blume's Designing Engineer. He was satisfied with the work.

Mr. C. M. Li (C.E.) was responsible for reviewing design verificati performed on the Turbine Building modifications. He verified that computer programs used on this project (DRAIN 2D and SAP IV) have be verified per the QA program; in addition, he reviewed the user file and test program certification. He reviewed a random 50% of the cal lations for the signatures of the designer and checker. He verified that design is reviewed by at least two other engineers against the design criteria. He checked the reviewer's calculations and feels that the results of the design review are acceptable. His contacts were H. Klice, D. Lang, and R. Yokoyama.

Mr. D. J. Curtis (M.E.) was responsible for verifying the quality of work on the piping support review. He chose Unit 1 fire piping supports at random. They were support Nos. 48/35R, 56N/54R, 555/6R, 58N/21R, and 41/57R.

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Calculations were checked to insure that the following items were properly included and calculated:

(a) Design Criteria,

(b) Consideration to Interfacing Disciplines (i.e., giving reaction loads),

(c) Consideration to Gaps (i.e., space required for thermal movement and maximum space allowed to be considered a seismic restraint),

(d) Natural Frequency,

(e) Actual Loads and Stresses, and

(f) DE Deflections.

It was not the intention of this review to investigate the constructability of modifications proposed by URS/Blume.

It was noted that, in the calculations for pipe support 56N/54R, a reducing factor for the allowable load on a concrete expansion anchor was used. PGandE pointed out that this was unnecessary and unduly conservative. URS/Blume acknowledged and stated that this practice is no longer being done. All other items were found to be satisfactory. URS/Blume is, in general, doing a satisfactory job of documenting the adequacy of pipe supports.

Mr. V. M. Chauhan (M.E.) was responsible for verifying the quality of work performed on the piping support review. He examined the certification of computer programs and the logic basis, reviewed hand calculations, analysis used in spectra programs, and justification for assumptions used in developing the analysis.

He found two errors where incorrect data were entered into the computer for computing hanger loads. Investigation revealed that the incorrect data resulted only in a more conservative load analysis.

He reviewed the "Rapid Reanalysis of Piping Systems - Hosgri 7.5 Evaluation" which is a program developed by Blume in 1977. The user manual is incomplete and contains no certification or verification. This engineer has several other yet unanswered questions which are documented on Supplier Audit Finding Reports Nos. 91605-11 and -12; resolution of these findings will be verified by Quality Assurance.

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Mr. R. E. Bachar (M.E.) was responsible for investigating the verification and certification of the PIPESD program. He reviewed t background information for versions 5.1, 5.2, and 5.3; all revisions and changes to this program have been verified through the use of benchmark problems. In three separate problems, PIPESD was compared respectively against ADLPIPE, ANSYS, PIPDYN, and SAP IV; in all case the solutions were acceptable and the analysis documented to the satisfaction of the responsible engineer.

It is felt that the critical review of Blume's performance on the Diablo Canyon Project has resulted in strengthening their program an providing our Engineering Department with sufficient data to verify the quality of work performed by Blume's engineers.

APPENDIX D

PERFORMANCE FACTORS FOR U. S. NUCLEAR PLANTS

JANUARY 1, 1979 TO MAY 31, 1982

APPENDIX D

PERFORMANCE FACTORS FOR U.S. NUCLEAR PLANTS

JANUARY 1, 1979 TO MAY 31, 1982 *

NAME OF PLANT	CAPACITY (MDC net)	DATE OF COMMERCIAL OPERATION	CAPACITY FACTOR (SINCE 1979)	AVAIL-ABILITY FACTOR (SINCE 1979)	FORCED OUTAGE RATE (SINCE 1979)
BWR					
Browns Ferry 1	1065	8/1/74	66.6	73.7	7.4
Browns Ferry 2	1065	3/1/75	75.2	82.0	7.3
Browns Ferry 3	1065	3/1/77	59.1	64.5	12.6
Brunswick 1	790	3/18/77	48.7	61.1	14.5
Brunswick 2	790	11/3/75	43.0	56.9	18.4
Cooper	764	7/1/74	65.5	78.5	2.7
Dresden 2	772	6/9/72	66.9	80.3	4.4
Dresden 3	773	11/16/71	57.5	70.7	13.6
Duane Arnold	515	2/1/75	59.8	74.9	12.5
Fitzpatrick	800	7/28/75	56.5	63.8	21.9
Hatch 1	717	12/31/75	55.4	63.1	16.4
Hatch 2	790	9/5/79	55.6	65.9	9.3
Millstone 1	654	3/1/71	62.4	79.7	9.8
Monticello	536	6/30/71	78.8	83.0	3.4
Nine Mile Pt.	610	12/1/69	65.4	61.6	9.4
Oyster Creek	620	12/1/69	51.3	58.1	28.4

* Data taken from NUREG-0020, Data For Operating Plants as of January, 1979 and May 31, 1982.

APPENDIX D

(Contd)

NAME OF PLANT	CAPACITY (MDC net)	DATE OF COMMERCIAL OPERATION	CAPACITY FACTOR (SINCE 1979)	AVAIL-ABILITY FACTOR (SINCE 1979)	FORCED OUTAGE RATE (SINCE 1979)
Peach Bottom 2	1051	7/5/74	65.2	70.0	8.3
Peach Bottom 3	1035	12/23/74	64.0	67.0	8.7
Pilgrim 1	670	12/1/72	59.6	65.8	9.0
Quad Cities 1	769	2/18/73	70.6	84.3	3.5
Quad Cities 2	769	3/10/73	57.1	73.9	7.5
Vermont Yankee	504	11/30/72	78.1	81.6	5.3
PWR					
Arkansas 1	836	12/19/74	54.1	64.0	21.2
Arkansas 2	912	3/26/80	57.8	69.3	21.3
Beaver Valley 1	800	10/1/76	28.0	32.9	40.9
Calvert Cliffs 1	810	5/8/75	70.2	75.4	9.1
Calvert Cliffs 2	810	4/1/77	82.9	85.4	5.6
Cook 1	1044	8/27/75	68.7	73.4	11.0
Cook 2	1082	7/1/78	68.8	71.5	16.3
Crystal River 3	797	3/13/77	55.2	59.8	22.0
Davis-Besse 1	906	11/20/77	39.5	56.1	22.5
Farley 1	829	12/1/77	41.5	47.4	25.3
Farley 2	814	7/30/81	84.5	88.5	11.5

APPENDIX D

(Contd)

NAME OF PLANT	CAPACITY (MDC net)	DATE OF COMMERCIAL OPERATION	CAPACITY FACTOR (SINCE 1979)	AVAIL-ABILITY FACTOR (SINCE 1979)	FORCED OUTAGE RATE (SINCE 1979)
Fort Calhoun	457	9/26/73	79.5	78.7	1.9
Ginna	470	7/1/70	68.8	69.9	4.4
Haddam Neck	550	1/1/68	83.0	83.9	2.0
Indian Point 2	864	8/1/73	57.3	64.1	12.1
Indian Point 3	965	8/30/76	42.7	59.2	20.6
Kewaunee	517	6/1/74	78.2	81.1	5.5
Maine Yankee	810	12/8/72	67.8	74.1	12.8
McGuire 1	1180	12/1/81	34.5	69.9	30.1
Millstone 2	864**	12/26/75	65.1**	68.6	16.4
North Anna 1	898	4/5/78	62.7	72.6	8.3
North Anna 2	890	12/14/80	62.9	67.6	17.1
Oconee 1	860	7/15/73	56.3	61.1	23.2
Oconee 2	860	9/9/74	58.9	67.7	12.4
Oconee 3	860	12/16/74	60.4	63.5	21.1
Palisades	635	12/31/71	52.1	50.4	26.0
Point Beach 1	495	12/21/70	63.5	78.9	3.3
Point Beach 2	495	10/1/72	82.7	86.1	1.4

APPENDIX D

(Contd)

NAME OF PLANT	CAPACITY (MDC net)	DATE OF COMMERCIAL OPERATION	CAPACITY FACTOR (SINCE 1979)	AVAIL-ABILITY FACTOR (SINCE 1979)	FORCED OUTAGE RATE (SINCE 1979)
Prairie Island 1	507	12/16/73	76.8	82.5	6.6
Prairie Island 2	507	12/21/74	82.4	85.9	4.7
Rancho Seco 1	873	4/17/75	54.5	63.4	20.4
Robinson 2	665	3/7/71	57.6	65.0	15.1
Salem 1	1079	6/30/77	45.7	53.9	27.6
Salem 2	1106	10/13/81	87.1	97.5	2.5
San Onofre 1	436	1/1/68	41.9	33.6	63.2
Sequoyah 1	1128	7/1/81	53.0	60.9	27.4
St. Lucie 1	777	12/21/76	75.9	76.3	4.3
Surry 1	775	12/22/72	40.1	57.2	38.6
Surry 2	775	5/1/73	43.0	46.7	8.6
Trojan	1080	5/20/76	60.6	68.2	8.2
Turkey Point 3	646***	12/14/72	45.1***	46.4	9.8
Turkey Point 4	666	9/7/73	72.7	75.8	3.1
Yankee-Rowe 1	175	7/1/61	55.9	64.7	23.7
Zion 1	1040	12/31/73	61.6	68.4	17.5
Zion 2	1040	9/17/74	54.6	68.1	21.0

** Licensed Maximum Dependable Capacity for Millstone 2 was 810 MWe until June 27, 1979. The capacity factor for Millstone 2 was calculated using a time-weighted average.

*** Licensed MDC for Turkey Point 3 was 666 until August 1980. The capacity factor was calculated using a time-weighted average.

THE FUTURE OF CONVENTIONAL NUCLEAR POWER: NUCLEAR REACTOR REGULATION

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Submitted to:

Energy Program
Office of Technology Assessment
Congress of the United States
Washington, D.C. 20003

Under Contract No.: 233-8400.0

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December 31, 1982

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PREFACE

This report, prepared for the Congressional Office of Technology Assessment, under Contract No.: 233-8400.0, supports an OTA study designed to identify and assess the impediments to the future of conventional nuclear power (light water-cooled reactors) in the United States. This particular document examines the topic of nuclear reactor regulation. The draft report was used as a background document for a workshop on Technological and Regulatory Changes held in Washington, D.C. on December 8 and 9, 1982. To the extent that they contributed to an understanding of regulatory impediments and possible solutions, the discussions held at that workshop are incorporated herein. It is not the purpose of this report to recommend or support specific proposals for regulatory reform, but rather to identify and discuss the important issues in the consideration of reform.

EXECUTIVE SUMMARY

The nuclear power industry is at a standstill, with the only new reactor order since 1976 on hold and record numbers of cancellations of plants well along in construction. Decreased load growth and exploding costs are clearly responsible, in part, for the moribund status of the industry. But uncertainty -- not only in costs and demand, but also in the licensing process and regulations governing the construction and operation of nuclear power plants -- is perceived to be the principal obstacle to the revitalization of the industry.

The 12 to 15 years it now takes to bring a nuclear plant on line represents an increase of more than 70 percent in the average lead time for nuclear units that began operating during the first half of the 1970s. The average construction period for plants completed in the 1978-82 time frame is 2.6 times that for plants completed in 1969 and 1970. What are the reasons for these substantial increases in lead time and construction period? Three case studies were performed in an attempt to answer this question. A deliberate attempt was made to select a licensing history perceived to be typical (construction period approximately nine years), another perceived to be unusually trouble free (construction period approximately seven years), and a third perceived to be exceedingly difficult (construction period approximately 10 years). All three plants examined in the case studies were completed, or are expected to be completed in the 1981-83 time frame. The primary sources of information were the licensing project managers employed by the licensees.

The case studies confirm industry contentions that new and/or changing regulatory requirements and standards have significantly lengthened construction time over the past decade. Lessons learned from operating reactors accounted for a significant fraction of the new regulatory requirements. Also, field construction problems stemming from what may be ill-conceived seismic criteria can be blamed for another sizeable fraction of the schedule slippages. However, instability in the regulatory process is not the only reason for lengthened construction periods. Deliberate slowdowns because of decreasing load forecasts also played a significant role, as well as management and coordination problems. Additionally, there is evidence that

the rapid growth in utility, A/E, vendor, and Nuclear Regulatory Commission (NRC) staffs in the mid-1970s took its toll by confounding the decision-making process.

The case studies also suggest that the difference between relatively difficult and relatively trouble-free licensing histories may be attributable, in part, to the experience and commitment of management personnel within the utility and at the construction site. Constant and aggressive interface with the NRC staff by utility management personnel may also play a significant role.

A cursory examination of reactor types other than the conventional LWR suggests that a significantly shortened licensing process would not result unless the risk of major fission product release can be shown to be several orders of magnitude lower than that of the LWR. These reactor types include High Temperature Gas-Cooled Reactors (HTGRs), Canadian (CANDU) reactors, small LWRs, and reactor design concepts explored in the early development phase of nuclear power. Moreover, with the exception of smaller or redesigned LWRs, no commercial fuel cycle exists in the United States for most of these other reactor types; even if it did exist, it would take a number of years to put into place another regulatory regime. So-called "ultra-safe" reactors, for which by definition the risk of major fission product release is orders of magnitude lower than conventional LWRs, should not be susceptible to backfit requirements, so that it can be argued that it would be relatively easy to license such a reactor. However, "ultra-safe" reactors (i.e., the ASEA-ATOM PIUS design) exist only on paper.

A number of licensing reform proposals have been drafted by the nuclear industry [the Atomic Industrial Forum (AIF)], the Department of Energy (DOE), and the NRC. Some or all of these proposals will be introduced as legislative packages into the 98th Congress. All of the proposals contain provisions for early site approval and preapproval of designs. These preapproved sites and designs would be valid for a number of years and would be stabilized by strong backfitting provisions. Some of the proposals would even allow for preapproval of site "characteristics" and design subsystems.

All of the proposals also contain backfit reform provisions, although the NRC proposes to accomplish backfit reform administratively. There is some disagreement on how to stabilize the regulatory process. DOE recommends that all backfits be subjected to centralized review and approval and that a backfit may be approved only if it is needed to reduce the risk of the overall plant operation to an acceptable level. AIF takes issue with the DOE language relating to an acceptable level of risk, preferring instead that the value of the added protection provided by the backfit exceeds the increased costs. The language of the NRC draft rule requires that a backfit provide a "substantial" increase in plant safety as well as a benefit which exceeds costs.

One-step licensing is also a provision in all of the reform packages. Although the greatest benefit from this reform would accrue to the utility who arranges a "marriage" between a preapproved site and a preapproved design, all of the reform packages permit the one-step option for all applications. All of the packages also propose a hybrid hearing process, which combines elements of both legislative and adjudicatory hearings. The NRC proposal also includes a number of administrative reforms aimed at tightening up the hearings process.

Other proposed legislative changes include:

- Relieve the Advisory Committee on Reactor Safeguards of its mandatory review responsibilities for each license application.
- Permit an authorized Federal, regional, or State government to certify the Need for Power.
- Allow communication between the Commission and other elements of the NRC during a licensing determination (i.e., ease the "ex parte" rule). A related proposal would bring the Commissioners fully into licensing decisions by making the Atomic Safety and Licensing Appeal Board a Commission staff office.
- Increase the use of rulemaking to resolve generic issues that affect several licenses.
- Eliminate the existing provision that provides for expiration of the Construction Permit if the facility is not completed within a designated period of time.

These proposals would appear to encounter little resistance from the nuclear industry, the NRC, or the intervenors.

A number of additional proposals for reform of the NRC and its regulations have been brought up from time to time. These include:

- Transfer the anti-trust review and export license responsibilities to the Departments of Justice and State.
- Transfer all or some of the NEPA responsibilities to the regional or State governments.
- Increase public participation during staff review by holding meetings in the vicinity of the site and soliciting public involvement.
- Decentralize the licensing authority by shifting some of the NRC's responsibilities to the Institute for Nuclear Power Operations.
- Replace the five-member Commission with a single administrator.
- Revise the entire fabric of the regulations using probabilistic risk analysis as a tool, aiming toward quantitative safety goals.
- Consolidate the entire Agency at a single location.
- Fund the intervenors.

Although none of the above is included in the current draft reform proposals, some or all of these changes could potentially accomplish more to reform nuclear regulation than the draft reform proposals.

It can be argued that several of the recommended regulatory reforms are either already in place or can be accomplished administratively without the need for legislation. Moves toward centralized review and safety goals have been initiated within the NRC. Provisions of the existing regulations already allow NRC to review site suitability and approve standardized plant designs. These decisions, however, are currently made outside of the process for considering an application for a license to construct a production and utilization facility, and are thus not binding on that process. The issue, then, boils down to stability.

In considering regulatory stability, it is important to view the NRC and the regulatory process as it is today, not as it was in the previous decade. An argument can be made that the process has already stabilized. The flood of industry standards and regulatory issuances appears to have subsided. A well documented, debugged Standardized Review Plan is in place. Hundreds of plant-years of operating experience have been amassed with large LWRs and

mechanisms for assessing operating experience have already been put into place. Only the large number of outstanding safety issues, looming in the background, gives pause to the perception of stabilization of the licensing process.

The use of safety goals or cost-effectiveness prescriptions to stabilize the regulatory process carries attendant risks. If the existing process, based on technological standards, has indeed stabilized, the change to a performance standard could move the licensing process in a totally new direction. Also, although probabilistic risk analysis is a useful tool for systematically examining systems and comparing alternative designs, the results of using this relatively new (for the nuclear industry) tool are potentially so highly uncertain as to negate its usefulness.

The issue in one-step licensing involves the finality of the design. Although there is little question of the efficacy of one-step licensing for preapproved sites and designs, for a custom plant the process would probably take longer than the current Construction Permit review. To what extent the design would have to be "set in concrete" and how much of the actual hardware would have to be specified before the NRC staff would be in a position to accept the design is an unanswered question.

Standardization has been proposed as a cure-all for the ills of the nuclear industry for at least a decade. All of the major nuclear steam supply system vendors have submitted standardized designs to the NRC for approval. Still, every operating plant and every plant under construction are essentially unique. Many knowledgeable critics argue that to accomplish the desired degree of industry standardization would require the plant to be standard down to the last nut and bolt, and possibly to have been carried all the way through licensing and operation before it would qualify. It is not likely that the industry or the NRC would take the necessary steps toward actual standardization voluntarily.

Discussion of the hearings process evokes the strongest emotions on all sides of the nuclear power controversy. DOE contractors estimate less than three months savings in lead time from the proposed hybrid hearings reform. This

may be a small price to pay if the existing process permits adequate public involvement and results in public confidence in the technology.

Without question, public confidence is necessary if there is to be a future for nuclear power. To achieve public confidence, accident-free operation of existing plants must be accrued. Until such time as the political issue surrounding the acceptability of nuclear power is resolved, an efficient hearings format may not be achievable.

1.0 INTRODUCTION

1.1 Status of the Nuclear Industry

It is generally accepted that the nuclear power industry is at a standstill. Since November of 1978, when the Tennessee Valley Authority received a construction permit (CP) for the Yellow Creek Station, no CPs have been issued. No new plants have been ordered or announced since 1978, before Three Mile Island (TMI), and the only order since 1976 is on hold. From 1978 through 1981, 44 units were cancelled.* Of course, plants had been cancelled before 1978, but the annual number of cancellations in recent years is about 30 percent higher than yearly cancellations in the 1974-77 period.[1]

There are a few basic and interrelated reasons for the moribund state of the industry. The first is cost: the real dollar cost of building a nuclear plant has risen almost continuously since the construction of the first competitively priced plant.** Costs for constructing coal-fired stations also have increased, but the rate of increase for fossil-fired units has lagged behind that of nuclear plants. The second factor is decreased load growth: after the oil embargo in 1973, load forecasts dropped dramatically from the 7 percent annual growth era. Over the past year or two, in fact, the nationwide demand for electricity actually has fallen. Decreased load growth has affected fossil-fired plants as well; in 1981, six fossil plants were cancelled and 36 fossil generating units were deferred by a year or more. Scheduled additions of fossil units drop steadily, from 23 units in 1982 to 11 in 1986.[2]

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*Currently, approximately 80 reactors are in operation, producing nearly 13 percent of the nation's electricity.

**This plant, Oyster Creek, was completed in 1969 and was a turnkey job; the actual costs may not have been competitive.

The third factor is uncertainty: uncertainty in costs, demand, and regulations. The 12 to 15 year lead time required for a nuclear unit is anathema to utility planning, particularly in these times of uncertain load growth. How much of this lead time and, for that matter, the increase in costs, are due to regulation will be discussed later.

1.2 The Current Licensing Process

As background against which to examine proposed licensing changes, we briefly outline the process in place now. The current Nuclear Regulatory Commission (NRC) licensing process involves, as it always has, two separate stages. A company wishing to build a plant at a particular site must first obtain a construction permit (CP).* The CP application includes a Preliminary Safety Analysis Report (PSAR), which presents the preliminary plant design, proposed safety features, and data relevant to safety considerations at the proposed site; an Environmental Report (ER) that is the basis for evaluating the environmental impact of the proposed plant; and anti-trust information. The NRC staff reviews the submittals to determine if they are complete enough to allow a detailed review, and requests additional information, if needed. The application is formally "docketed" when it meets the minimum acceptance criteria.

Over the next one to two years, the NRC Office of Nuclear Reactor Regulation (NRR) compares the details of the application with the Standard Review Plan (NUREG-0800), asking the applicant to provide more data if needed. The staff then prepares a Safety Evaluation Report (SER) documenting the review. Concurrently, the Advisory Committee on Reactor Safeguards (ACRS) also reviews and comments on the application, and the staff may issue supplements to the SER in response to issues raised by the ACRS.

Either concurrently with or separately from the safety review, the staff reviews the proposed plant's environmental impact and prepares a Draft Environmental Impact Statement (DEIS). The DEIS is issued for review by

*Before applying for a CP, a utility typically spends two to three years selecting a site, conducting environmental baseline studies, and preparing the supporting documentation for the application.

involved state, Federal, and local agencies and interested private parties. After their comments are received and questions are resolved, a Final Environmental Impact Statement (FEIS) is prepared and issued.

A public hearing is required before the CP can be issued. The hearing is conducted by a three member Atomic Safety and Licensing Board (ASLB), and members of the public may participate. Both environmental and safety issues may be considered, or two separate hearings can be held. The SER and FEIS (and any supplements) are the major pieces of "evidence" in this adjudicatory hearing.

If safety and environmental issues receive a favorable decision by the ASLB, a construction permit is issued to the applicant by the Director of Nuclear Reactor Regulation. This initial decision can be reviewed by an Atomic Safety and Licensing Appeal Board (ASLAB) and by the Commission itself. Under certain circumstances, a Limited Work Authorization (LWA) can be issued, which allows a restricted amount of work to be done onsite before the CP is issued. Antitrust hearings are held separately from safety and environmental hearings.

Obtaining an operating license (OL) is the second stage of the process, and the various steps closely parallel those of the CP stage. When construction has progressed to the point where final plant design and operating plans are sufficiently complete, the applicant submits a Final Safety Analysis Report (FSAR). The NRC/NRR staff again reviews this new submittal, asks for additional information, and prepares another SER for the operating license. The ACRS also reviews the submittal independently and reports to the Commission. The major difference under law is that a public hearing is not mandatory at the OL stage, though one is held if, after the Commission publishes a notice that it is considering the issuance of an OL, an affected party petitions the NRC for a hearing. If a hearing is held, the same adjudicatory and decision process pertains as for a CP.

1.3 Regulation

The 12 to 15 years it now takes to bring a nuclear plant on line represents an increase of more than 70 percent in the average lead time for nuclear units

that began operating during the first half of the 1970s.[1] The time it takes to bring a nuclear plant into commercial operation can be broken down roughly into two periods. The first is the time from the decision to build the plant to the close of the CP hearing. This includes the utility planning phase, the NRC review of the application, the preparation of the SER and EISs, and the CP hearing. The trend for this time period is given in Table 1 (the utility planning phase is not included because it varies substantially).

Table 1

AVERAGE TIME TO OBTAIN A CP[3]

(Months after docketing)

Date of Commercial Operation	Months
1959-63:	11
1967-68:	8
1969-70:	20
1971-73:	33
1973-75:	21
1975-78:	26
1978-82:	30
1983-85:	41

The passage of the National Environmental Policy Act (NEPA) in 1969 apparently caused a substantial increase in CP lead time. NEPA's impact, however, was probably not felt fully until the last half of 1971, after the Calvert Cliffs decision. Regulatory review time also increased after Schlesinger's arrival at about the same time, due to increases in the number of new applications, more sophisticated SARs, and the growing number and complexity of codes, guides, and standards. The change from project review to disciplinary matrix review within the NRC [Atomic Energy Commission (AEC) until 1974] in the early 1970s also probably contributed to the increase in review time.

The second time period involves construction itself--the time between issuance of the CP and the OL. This by far is the period of major utility investment, and most critics feel that delays here can increase costs substantially. The trend for this time period is shown in Table 2.

Table 2

AVERAGE CONSTRUCTION TIMES IN MONTHS[3]

OL Issued	Months
1959-63:	32
1967-68:	44
1969-70:	43 (excludes 11 months LWA activity)
1971-73:	45 (excludes 11 months LWA activity)
1973-75:	60 (excludes 8 months LWA activity)
1975-78:	69 (excludes 9 months LWA activity)
1978-82:	112 (includes 18 months TMI delay and excludes 5 months LWA activity)
1983-85:	103 (includes 18 months TMI delay and excludes 8 months LWA activity)

The average construction period for plants completed in the 1978-82 time frame is 2.6 times that for plants completed in 1969 and 1970, including delays attributable to TMI; without TMI delays, it is nearly 2.2 times as long.

A number of factors have contributed to lengthening construction periods and higher costs. More commodities--concrete, piping, electrical cable--are required; physical plant designs are more complicated; and worker productivity has dropped substantially, at least in part because plants are harder to build.[4] FSAR review and hearings take place during roughly the last three years of the construction period, and thus do not contribute directly to delay. However, there is no doubt that increases in regulatory requirements lead to higher commodity use and to more design complexity. The explosion in numbers of regulatory guides and industry standards during the 1970s are in no small part to blame. Seismic requirements are a prime example. Delays resulting from moving regulatory targets also contribute to declining labor productivity, because rework is never as satisfying as original construction and because crowding from support structures makes construction access difficult.

Regulatory escalation is perhaps best typified by the NRC reaction after the accident at Three Mile Island. Licensing virtually halted as the Commission reacted to the accident, and the numbers of I&E (Inspection and Enforcement) Bulletins and Circulars increased dramatically as new requirements were issued. In 1979 and 1980, the number of I&E Bulletins more than tripled, and the number of I&E Circulars more than doubled. By the end of 1979, approxi-

mately 34 I&E Information Notices had been issued; by the close of 1980, 82. At the end of 1980, a total of 257 information/action documents had been issued--almost four times the number in circulation at the close of 1978.

However, an overriding issue is whether regulation has increased costs unnecessarily--beyond the public's demand for safety, in such a way as to increase costs but not safety, or by inefficiencies that produced unnecessary delays. The evidence suggests that in the 1966-1972 period, regulatory requirements fell short of public demand for safety. A similar "safety gap" was perceived after TMI. Of course, the public has consistently perceived the risk from nuclear accidents to be higher than the risks estimated by the experts. Whether safety was (or is) actually served by the existing regulatory process is, unfortunately, open to debate.

Another complicating factor--whether procedural delays in fact actually influence plant costs--is germane. A Rand Corporation statistical analysis[5] indicated in 1978 that whereas nuclear power plant capital costs were strongly correlated with the time of CP issuance (showing an increase of $117/kw per year), there was no correlation between capital costs and length of construction, once the effect of CP issuance date was separated out. Karl Cohen, in recent correspondence with the NRC,[6] has also asserted that "it is not true that shortening the construction period will substantially reduce nuclear plant costs... ." If these assertions are true, the explanation is apparently that somehow utilities can cope fiscally with construction delays.

2.0 CASE STUDIES

In order to pursue the issue of the impact of regulation on reactor lead time, three licensing case studies were conducted. A deliberate attempt was made to select a history perceived to be typical, another perceived to be unusually trouble free, and a third perceived to be exceedingly difficult. All three plants examined in the case studies were completed, or are expected to be completed in the 1981-1983 time frame. The following writeups of these case studies were performed by SC&A, and they represent our understanding of the data provided by a number of sources. The primary sources of data were the licensing project managers employed by the licensees. Information was also obtained, insofar as possible, from NRC staff and former staff members. A more thorough investigation would have solicited input from the utility/A-E design staff members, construction project management, and utility planning groups. However, time and resources for these case studies were limited. Nevertheless, SC&A assumes responsibility for any omissions or misinterpretations.

2.1 Case Study #1

The utility began planning for this two-unit station in the late 1960s. This case study focuses on the first unit, for which the NSSS was ordered in 12/69. The PSAR was filed in 9/70, followed by an ER in 3/71 (a short document in those pre-Calvert Cliffs' days). There was no acceptance review in those days, and the utility held an informal meeting with the NRC (AEC) staff in 12/70. The first round of formal questions was received by the utility in 2/71. Another round was received in 4/71.

The anti-trust review, which was still in progress at the time of the issuance of the CP, was uneventful. The review, which was handled "off to the side by the lawyers," resulted in an amendment to the CP requiring the utility to supply bulk power to neighboring entities.

The staff issued the SER in 12/71, the DEIS in 7/72, and the FEIS in 10/72. There were two major contentions: the design of a pond to be used as an ultimate heat sink, and the ice condenser used to suppress pressure in the

containment in the event of a major steam release. Only one other plant had proposed an ice condenser at that time, and there were problems with that earlier proposal. The staff review resulted in some design changes (higher in ice loading), which may have added an additional margin of safety. The design of the pond was not altered.

The meeting with the ACRS was held in 9/71 and the ACRS letter went to the Commission in 10/71. The ACRS review also focused on certain unique aspects of the containment resulting from the ice condenser design. The upshot of the review was a modification to the design in which safety-related instruments and design interlocks were added. The ACRS review identified a real safety problem, albeit one involving very low probabilities.

An organized intervenor group, comprised largely of one individual, revealed its existence at the time that the plant was announced, and filed for participation in the hearings. The first prehearing conference was held in 2/72, discovery began in 3/72, and the radiological hearing convened for a few days in 6/72. The radiological hearing reconvened in 9/72 and 10/72, for a total radiological hearing time of approximately three weeks. A three-day environmental hearing was held in 11/72. The ASLB rendered its decision in 2/73, and the CP was issued three days later. The intervenors appealed the hearings to the ASLAP and filed a number of exceptions.

The major contentions of the intervenors involved the ice condenser and the reactor vessel studs. The preliminary design, however, was not changed as a result of these contentions.

Although the CP was not issued until 2/73, the utility had begun certain work under an "exemption" (in those days, similar to a Limited Work Authorizations). Construction of the pond dam began in 6/71 and foundation work for the containment and auxiliary buildings started in 1/72. In fact, the target date for fuel loading at that time was 5/75.

However, not long after the CP was issued, the utility began to revise the schedule. Formal submissions to extend the schedule (required because under

the existing law the CP expires after a certain period of time) were filed several times through 1976. The reasons for the extensions were:

- Delays in getting started on foundations
- Equipment delivery delays
- Material delivery delays
- Delays in receiving drawings
- Lack of craft manpower
- Lack of design information
- Coordination problems
- Financial considerations
- Load growth considerations

In fact, load growth considerations may have played a very significant role in the pace of construction during those early years. After the oil embargo, demand began to fall throughout the utility industry, and this utility appeared to have adequate reserve margins (20-30 percent) into 1979. In 11/76, the utility delayed fuel loading until 10/78, for commercial operation early in 1979.

During those early years of construction, it also became apparent that the original schedule was very optimistic. Real problems in project management and coordination began to plague the project. The original schedule was just too optimistic. The staff was growing and the organization was changing, and coordination between the players was becoming more difficult. The vendor had several other similar plants to consider when a decision on this plant was needed, and therefore vendor decisions took longer to make.

Regulatory impact on the construction schedule was not apparent until late 1976 or early 1977. In fact, an event in early 1976 should have been recognized as a precursor of considerable significance. The utility began to have problems at that time with the designer/supplier of some of the seismic supports, but it was not recognized until 1977 that these problems might eventually have serious repercussions on the design schedule. In 7/77 the utility requested a six month delay in the schedule to approximately 4/79, and

cited the cancellation of the seismic support contract, as well as a reevaluation of the hanger design, as the primary reasons for the delay.

In fact, the seismic support contractor was not solely responsible for piping design problems that began to affect the construction schedule at that time. As early as 1975 (in response to the FSAR, discussed below) the NRC expressed concerns about pipe whip and pipe breaks. Until that time, no one had considered the impact of safety-related piping running through non-safety-related components. Unfortunately, most of the equipment at that time had already been specified and some equipment was in place. Considerable reanchoring and redesign resulted from this safety concern. Also, some new equipment had to be ordered.

Seismic support problems had the greatest effect on construction time in the 1978 - 1979 time frame, when the utility quality assurance (QA) staff discovered that there were deficiencies in hanger inspection. The QA staff found that the previous inspections were not capable of determining if the hangers had actually been installed according to design. Thus all 14,000 hangers had to be reinspected, resulting in considerable redesign and reinstallation. The rework was exacerbated by a relatively congested containment, a result of the ice condenser design. This entire exercise cost the project at least a year of construction time.

Other backfit exercises during the protracted construction period also affected the schedule. It is difficult to assign a specific impact on the schedule to any one of these issues, but taken in the aggregate, there had to be an effect.

The physical security requirements were promulgated in 1976 (Part 73.55). As a result, a physical security system at a cost of approximately $4M had to be installed.

As a result of the Brown's Ferry fire, the NRC required the utility to look at the fire protection design, including cables and penetrations, and to submit a document supporting the evaluation. Quite a few cables had to be moved, but the major backfit was the installation of a $5M standby shutdown facility.

Analysis for the fire protection exercise was largely completed in 1977, with construction starting in 1978, and continuing through 1979 and 1980.

In 1977, the vendor uncovered a design problem in the auxiliary feedwater system, requiring four new containment penetrations and hundreds of feet of piping. In addition, steam generator problems began to appear in this same time period, resulting in repairs costing as much as $10M.

Then in 1979 the Three Mile Island accident occurred, resulting in many small backfits--new cable, new radiation monitors, new procedures, redesign of the containment isolation valve, etc. For the most part, these changes were absorbed in the other larger delays. However, if these other delays had not been present, TMI would no doubt have been on the critical path.

TMI did not have as large a perceived impact as might have been expected because another problem occurred in 1979. At that time, the plant was 90 percent complete and hot functional testing had begun. Again, the pipe hangers were the problem, resulting in a lot of rework. Apparently, thermal expansion of piping revealed many problems resulting from the close tolerances specified by the seismic support design. Another hot functional test toward the end of the year revealed more rework, resulting in another year's delay in the schedule--nearly the last delay.

Meanwhile, the licensing process proceeded as if a 1977 - 1979 schedule for fuel loading was realistic. The utility submitted the FSAR in 5/74 and the staff accepted it one month later. There were two rounds of formal questions, in 4/75 and 9/75, and thereafter they kept trickling in on an informal basis. In fact, there were 16 amendments to the CP, most resulting from informal meetings with the staff.

The staff OL review engendered quite a bit of analysis and several small changes in the plant design, some resulting in backfit. As discussed earlier, the staff's questions relating to the interaction between pipebreaks and seismic support resulted in considerable effort. This probably increased the margin of safety, but only for very low probability events. Also, the staff review required the utility to add a low pressure overpressurization

protection system in 1977. There were several small changes in the ventilation system. Some problems, such as pipe break, were handled by increasing the inspection frequency written into procedures. The staff still had a lot of questions about the ice condenser design, but they required no design changes. The emergency core cooling system (ECCS) also demanded a great deal of analysis.

The staff published the SER in 3/78, and it contained a few outstanding issues. The DEIS had already been completed in 12/75. The ACRS met in full committee on the OL in 4/78. There were no major contentions, and their letter went to the Commission during the same month.

As early as 1975, meetings were held with the intervenors (the same group that intervened during the CP proceedings) to agree on contentions for adjudication during environmental hearings. The environmental hearings were held for two weeks in 3/77 and one week in 4/77. Need For Power was the main contention.

The same intervenors also called safety hearings which began in 6/78, to contest financial qualifications plus a number of potential safety problems. Reconvened in 8/78, the hearings were completed with no decision. Then in 4/79, the ASLB rendered a positive decision on the OL, subject to some generic issues. In 5/80, with completion of construction beginning to appear as a reality, the utility asked the ASLB to lift the objections, and at about the same time, the intervenors petitioned the Board to reopen the hearings to examine the impact of a hydrogen burn on this low pressure containment. The petition was granted in 11/80 and the Commission granted a zero power OL in 1/81. In 3/81, three to four weeks of hearings were held and in 5/81 the ASLB gave permission for operation at power. A five percent power amendment to the license was granted in 6/81 and a full-power license in 7/81.

In fact, the OL hearings did not delay the full power operation of the plant. However, none of the contentions at the initial OL hearings or the later session resulted in any change to the design or planned operation of the plant. Despite the issuance of the full-power license in 7/81, the plant did not achieve criticality until later in the year and did not reach substantial power until 1/82. Ironically, because of steam generator problems, the plant had not been taken above 50 percent power as of 10/82.

Table 2

CHRONOLOGY: CASE STUDY #1

NSSS ordered	12/69
PSAR filed	9/70
Informal NRC/utility meeting	12/70
First round of questions	2/71
ER filed	3/71
Second round of questions	4/71
Pond dam construction initiated	6/71
ACRS meeting	9/71
ACRS letter to Commission	10/71
SER issued	12/71
Foundation work initiated	1/72
First prehearing conference	2/72
Discovery initiated	3/72
Safety hearing convened	6/72
DEIS issued	7/72
Safety hearing reconvened	9/72, 10/72
FEIS issued	10/72
Environmental hearing	11/72
ASLB decision	2/73
CP issued	2/73
FSAR submitted	5/74
FSAR accepted	6/74
First round of questions	4/75
Second round of questions	9/75
DEIS issued	12/75
Environmental hearings	3/77, 4/77
SER issued	3/78
ACRS meeting	4/78
Safety hearings	6/78, 8/78
ASLB decision with objections	4/79
Petition to lift objections	5/80
Petition to reopen hearings	11/80
Zero-power OL granted	1/81
Hearings reconvened	3/81
5% OL granted	6/81
Full power OL granted	7/81

2.2 Case Study #2

This case study describes the licensing process for a three-unit plant. The first unit expects to load fuel in mid-1983. All of the completely identical units should be operational by 1986.

The utility began planning for this nuclear station in 1970. A report of the recommendations was issued to the Participants in 4/71, and they decided to proceed with the project. The Participants in the project currently include five utilities, one of which was delegated responsibility to construct and operate the station and is the applicant and licensee with regard to the NRC licensing process. The Participants in the project have changed over time, resulting in some additional attention during the anti-trust review. None of these changes affected the project schedule, however.

A major first step taken by the applicant on behalf of the Participants was to hire and designate as Project Manager an individual with substantial nuclear plant design and construction project management experience. The applicant assembled a project staff and began negotiations to select the Architect-Engineer (A/E) and Nuclear Steam System Supplier (NSSS) for the project. Preliminary design work was started in order to prepare the PSAR, which was filed with the NRC in 6/74. Following a successful mini-review, the NRC docketed the CP application in 10/74.

The NRC's review of the PSAR and ER resulted in several rounds of questions, mainly to clarify design criteria and the applicant's positions with respect to NRC guidance documents. The anti-trust review was expeditious, considering the fact that some of the Participants in the project had changed.

The NRC issued the Draft and Final Environmental Statements in 4/75 and 9/75, respectively. The SER was issued in 10/75. There were no major contentions with the NRC staff at the CP stage and no significant changes were made in the preliminary design. This lack of contention was due in part to the fact that the applicant had established project design criteria which addressed and generally complied with the NRC's regulations and regulatory guidance in force at the time. In addition, the applicant made a concerted effort to anticipate

NRC requirements, which were burgeoning during the early to middle 1970s. To that extent, this project had an advantage over earlier projects in being able to establish plant design criteria that complied with developing NRC requirements.

The first of only two meetings with the ACRS was held in 11/75. There were no major contentions with the ACRS and a "clean" letter was issued within the month.

The ASLB hearings began in 2/76 and lasted only five days. Minor organized intervention was abandoned prior to the ASLB hearings. No major contentions were raised during the ASLB hearings and a favorable decision was issued in 5/76. However, the ASLAB did review the decision since the project involved a standard NSSS design, a relatively new concept in 1975.*

The CP for the three-unit station was issued in 5/76, approximately twenty-three months from initial filing and nineteen months after docketing. Construction work is still going on with fuel loading at Unit 1 expected in 8/83.

The initial projected cost of the three-unit project was estimated in 1974 to be about $3.2 billion. The current projection is $4.3 billion, with Unit 1

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*The initial Commission policy statement on standardization of nuclear power plants was issued in 4/72. In a subsequent statement issued in 3/73, the Commission announced its intent to implement a standardization policy for nuclear power plants. WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," was issued in 8/74. Amendment 1 to WASH-1341, dealing with "options" and "overlaps," was issued in 1/75. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to Part 50 and Section 2.110 of Part 2 of 10 CFR.

The NSSS standard design for the case study project was submitted by the NSSS in the form of an application for a Final Design Approval from the Commission and was in response to Option 1 of the Commission's standardization policy. Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a design outside the context of a license application. The application for the NSSS's standard design was docketed in 12/73. The SER was issued in 2/75.

more than 96 percent complete, and Units 2 and 3 about 88 and 43 percent complete, respectively. The projected fuel load dates for Units 1 and 2 have slipped about nine months since the original schedule was announced in 1974.

The case study project has experienced some union problems, material supply problems, rework delays and other non-regulatory problems during construction. The utility took innovative steps early in the project, however, to ensure that such problems, which are typical of large construction efforts, would not substantially impede progress or affect quality. These measures included, for example, incentive/penalty contractual agreements with major contractors and suppliers, and unique no-strike bargaining labor agreements with construction labor unions. Additionally, the superior climate and site conditions and the availability of skilled labor have contributed to the construction progress.

Major rework efforts were associated with post-TMI requirements, particularly in areas related to fire protection and security. Many of the TMI requirements had been anticipated in the original plant design, however, lessening their impact substantially. The impact of these requirements on schedule has been relatively small.

The utility submitted the FSAR in 10/79 and it was docketed for review in 6/80. In a 3/80 meeting with the NRC staff, the applicant expressed its concern that the NRC staff review of the FSAR had not begun, even though the FSAR had been submitted in 10/79. The NRC (specifically NRR) was experiencing a severe shortage of personnel for OL casework due to TMI-related activities and the review of higher priority applicants. NRR indicated that a dedicated NRC staff review of the applicant's FSAR could not begin before 1/81. NRR proposed, however, to place NRC reviewers on the project, though not on a dedicated basis, if the applicant was willing to try new review techniques on an experimental basis. The technique (later know as Independent Design Reviews, or IDRs) consisted of an in-depth review of individual plant systems by the applicant which, if well documented and monitored by appropriate NRC staff personnel, could relieve much of the burden on the NRC staff to conduct such reviews on a question and answer basis. This process had been employed by NASA in the space program at the design development stage. The systems reviewed using the IDR approach were:

- Alternating current power systems
- Direct current power systems
- Auxiliary feedwater system
- Equipment qualification methodology
- Fire protection
- Instrumentation and control systems
- Auxiliary systems
- Containment systems

Another innovative approach used by NRR to expedite the licensing process for the case study project involved the "Draft SER Approach," or DSA. In the DSA, the staff eliminated the sets of questions (i.e., Q1s and Q2s) typically issued to the applicant for its response. Instead, a draft SER was prepared. The many open issues contained in the draft SER were resolved in a three to five day meeting held between key NRC, applicant, NSSS, and A/E personnel capable of reaching resolution on all issues. The meetings were open and technical personnel on both sides were able to discuss each issue in depth with supporting information (e.g., detailed analyses, drawings, vendor reports) available to assist in reaching decisions. The areas reviewed using the DSA were:

- Auxiliary systems
- Mechanical engineering
- Radiation protection program
- Instrumentation and control systems (follow-up to IDR)
- Structural engineering

The FSAR submitted for the case study project referenced the NSSS standard plant. Because of the timing of the FSAR and the standard NSSS submittals, the reviews were essentially conducted as an integrated review; i.e., the standard NSSS submittal was treated as an appendix to the FSAR.

The NRC issued the Draft and Final Environmental Statements at the OL stage in 10/81 and 2/82, respectively. The SER was issued in 11/81, just over two years after submittal of the FSAR by the applicant. There were some contentions with the staff at the OL stage but none which have or will significantly affect the project schedule.

The ACRS review time was minimal (initial meeting 11/8 with letter issued two weeks later) and there were no major contentions. The ASLB hearings began in 4/82 and lasted for 15 days over a three month period. No decision has been issued as of this date. There was (and continues to be) some organized intervention centered on issues related to the environment and the availability of cooling water. Both of these issues were reviewed fully at the CP stage and are not expected to delay receipt of the OL.

Table 4

CHRONOLOGY: CASE STUDY #2

Report to Participants	4/71
PSAR filed	6/74
CP application docketed	10/74
DEIS issued	4/75
FEIS issued	9/75
SER issued	10/75
ACRS meeting	11/75
ACRS letter to Commission	11/75
ASLB hearing	2/76
ASLB decision/CP issued	5/76
FSAR submitted	10/79
FSAR docketed	6/80
DEIS issued	10/81
SER issued	11/81
ACRS review	11/81
FEIS issued	2/82
ASLB hearings	4/82
Unit 1 expected fuel load	8/83

2.3 Case Study #3

This plant is the first nuclear unit built by this utility. Planning began in the early 1960s, though it was not until early 1967 that the utility solicited bids for the NSSS. The utility selected the AE/constructor in 5/67, and submitted the PSAR to the AEC one year later. Initially, the AE/constructor held major responsibility for construction management; utility control was not evident. No environmental report was necessary in this pre-NEPA era. The NRC staff had relatively few questions on the PSAR, and preliminary site work began under an exemption in 9/68. Anti-trust proved no problem.

Two months later, the utility placed a hold on the project that lasted about nine months, apparently because the NSSS vendor made a proposal to provide a higher megawatt system than originally planned. The utility accepted that proposal, and in 4/69 filed an amendment to the CP application. The amendment did not result in discussion of any new issues of substance, despite the fact that the utility did not propose to increase the size of the secondary containment. This decision later would result in equipment and workspace crowding, though that situation could not have been foreseen. However, the decision to scale up very probably delayed the review and CP hearing sufficiently to trigger the application of the Calvert Cliffs' decision.

In 8/69, site clearing and grading began, and by 2/70, the NRC staff and ACRS had completed their reviews, issued the SER, and published notice of the CP hearing. The utility began preliminary excavation and some construction in 3/70, and between 3/70 and 7/70, three prehearing conferences were held and the environmental report was filed (the short, pre-Calvert Cliffs' version). CP hearings began in 9/70, as did the foundation work for the turbine and reactor buildings.

In 1971 a number of things happened, some of which the utility could control and some of which it could not. Clearly not controllable were the AEC's reevaluation of the adequacy of emergency core cooling systems (begun in 5/71) and a decision in the Calvert Cliffs' case in 7/71. However, in 8/71, the utility voluntarily slowed engineering and essentially stopped its exempt construction work. This decision to halt was largely to avoid potential

rework stemming from the ECCS hearings, though other factors (see below) may have contributed. According to NRC representatives, there is some question as to whether the AEC would have required the utility to cease construction (they did so require early in 1972) if the utility had not shown a willingness to do so.

State environmental hearings also began in 8/71, and this, together with the requirement to prepare a full-scale environmental report, very likely was a factor in the decision to slow or stop engineering and construction. Intervenors were extremely active in both the AEC CP hearings and the state environmental hearings, and the Calvert Cliffs' decision gave the intervenors considerable new opportunity for further action in both hearings. The state environmental hearings were held over a four-month period, a short one when compared with the eventual multi-year hearing period for the AEC construction permit. Hearings on radiological issues began in 9/70 and concluded in 1/72. The utility filed its second (expanded) environmental report in 12/71. In 6/72, the AEC issued the DEIS, and engineering work speeded up. Comments on the DEIS were due to the AEC in 7/72; the CP was finally issued in 4/73. In total, this utility probably experienced, for its time, an extremely protracted CP hearing -- considerably more than 40 hearing days, though those days were spread out over more than a year. However, very little in the way of design or construction changes resulted from the new EIS process, the degree of intervention, or the relatively long hearings. It should be noted that the AEC issued no CPs from the end of 1971 to early 1973, due to the ECCS review and expanded EIS production.

After receiving the CP in 4/73, construction proceeded relatively smoothly for about two years. A few minor labor problems and work stoppages did occur, and some materials were not available when needed, apparently due to the imposition of the ASME code requirements for piping and valves. These factors did not cause substantial delay.

In 4/75, the NRC began to raise several generic issues, including those related to the Mark II containment and to pipe whip and restraint. The utility had to redo a great deal of preliminary engineering, and ultimately changed the support designs. It does not appear that the utility ever

formally slowed or stopped construction, but the emphasis of construction clearly shifted toward those areas that were less likely to require eventual rework. The Mark II containment issue almost certainly resulted in delay, at least some of it due to NRC reluctance to come to a decision, and the utility feels that risk was not substantially reduced. Issues of pipe hanger loads are still not completely resolved as of this writing (see below).

The utility submitted the FSAR in 9/75, and NRC conducted a mini-review that resulted in about 30 questions to the utility, most of them dealing with hydrologic issues. One concern, about the way the probable maximum hurricane (PMH) was analyzed, seems to represent the single regulatory issue unique to this plant. Until this time, NRC had not notified the utility that no documentation could be found that formally accepted the PMH analysis provided in the second environmental report. The utility was required to use a different computer model under quite severe water level assumptions. This reanalysis resulted in the utility rebuilding the cooling water intake and retaining walls.

In 1/76, the NRC formally docketed the OL application and began the review. At that time, the utility anticipated loading fuel in late 1978. However, to date, there have been over 800 questions on the FSAR. Intervenors have raised over 120 issues. The initial questions and answers were exchanged in 7/76, and the first prehearing conference was held in late 1976. The major FSAR contentions centered around PMH, Mark II, containment isolation, cable separations stemming from the Brown's Ferry fire, single failure criteria, and electrical isolation.

The large number of questions and the issues raised by at least four intervenors caused NRC to conduct a cautious and lengthy review. The reaction to the FSAR and the fairly continuous re-engineering required by NRC questions may also have encouraged the utility to take a much stronger role in construction management, beginning in 1977. Until that time, the AE/constructor had been in control, though the A/E did not even have a site office until 1975, roughly two years after the CP was issued. The A/E staff was also growing rapidly during the mid-1970s. NRC representatives feel that the tighter integration of the utility and the A/E and the assumption of

greater management control by the utility added about a year to the construction time, although it apparently made little improvement in construction scheduling.

Also continuing were the reanalyses of pipe stresses, first begun in 1975. While this problem was by no means unique to this utility, there was considerable discussion between NRC and the utility on the proper procedure for combining loads and then on precisely which physical equipment must be included in the analysis. In 1977, the utility moved the scheduled initial fuel loading into 1979, recognizing that the 1978 date would not be met.

All of these factors, plus a change in the NRC Licensing Project Manager in the spring of 1978, combined to push the NRC's review into the Three Mile Island era. NRC representatives estimate that, if they had published the SER before TMI, the licensing process could have been shortened by about 18 months. However, to assure that the OL hearings would go as smoothly as possible, they wanted to resolve as many issues as they could before publication. The NRC published the draft SER in 4/81, including responses to some of the issues raised by TMI. ACRS review turned up only minor issues, and the ACRS letter went to the Commission in the fall of 1981.

Pipe stress analyses were still continuing during the TMI hiatus. Agreement on how to combine loads was not reached until 1979, and in early 1980 the utility, in an attempt to anticipate the type of analysis NRC would desire, conducted a stress analysis for "worst-case" pipe hangers, submitting a Final Design Assessment Report in December of 1981. NRC rejected that analysis in August of this year, and asked for reanalysis.

OL hearings for this plant began in early May of this year and are still underway, at the rate of about two and one-half weeks of hearings per month. Intervenor groups have continued to be extremely active, raising issues of systems interaction, quality assurance, and emergency preparedness (emergency preparedness hearings begin in January of 1983). The county government was granted permission to intervene as a "neutral" party, and has demanded representation on the plant's nuclear review board and an independent review of all the plant's safety systems. The utility has acquiesced to the latter

demand, and is spending between two and three million dollars for the review. A particularly sticky issue with the county is the adequacy of the emergency plan, the local portion of which is the county's responsibility. The county has only recently (late November 1982) issued its portion of the plan, and there are considerable differences in scope from the plan the utility had proposed. At this point, the utility hopes to finish the primary containment momentarily, and expects to be in a position to load fuel in March or April of 1983. To do so, it may be necessary to obtain an OL under the provisions of the interim licensing legislation.

Table 5

CHRONOLOGY: CASE STUDY #3

NSSS bids solicited	early 1967
AE/Constructor selected	5/67
PSAR submitted	5/68
Preliminary site work	9/68
Project hold	11/68
Amendment to CP application filed	4/69
Site clearing and grading	8/69
NRC staff, ACRS review, SER issued, CP hearing notice	2/70
Preliminary excavation/construction	3/70
Prehearing conferences and Environmental Report filed	3/70 - 7/70
CP hearings	9/70 - 1/72 (>40 days)
Engineering slowdown/construction halt	8/71
State environmental hearings	8/71 - 12/71
Second Environmental Report filed	12/71
DEIS issued	6/72
CP issued	4/73
FSAR submitted	9/75
OL application docketed	1/76
First round of questions	7/76
Prehearing conference	late 1976
Increased utility role in construction management	mid 1977
Draft SER published	4/81
ACRS review/letter	9/81
Final Design Assessment Report on Pipe Hangers submitted	12/81
OL hearings begin	5/82 (ongoing: 2.5 weeks of hearing per month)
Pipe Hanger report rejected	8/82
Emergency Preparedness hearings	1/83
Planned fuel load	4/83

3.0 PROPOSALS FOR REFORM OF THE REACTOR LICENSING PROCESS

This discussion on reform of the reactor licensing process centers around the recent recommendations put forth by the Department of Energy (DOE)[1] and NRC.[7] The DOE proposal, unaccompanied by detailed legislative language, closely resembles recommendations put forth by the Atomic Industrial Forum (AIF) approximately a year earlier,[8,9] which did include detailed legislative language. The obvious differences between the two, largely in the area of backfitting, are brought out in the discussion of that provision. The NRC legislative proposal, the Nuclear Licensing Reform Act of 1983, represents a second round of Commission recommendations. The first, the Nuclear Standardization Act of 1982,[10] was circulated for comment earlier this year, but was then withdrawn. The new NRC package is quite similar in many respects to the DOE and AIF proposals. The original NRC proposal was criticized by the nuclear industry for having too narrow a scope and opposed by the intervenors for changing the existing hearings and backfit procedures.[11]

3.1 Early Site Approval

At present, the NRC review of site suitability is not initiated until the CP application is docketed, and thus the review is on the critical path for the project. Indeed, substantial investment has generally been made by the utility in conceptual design and other project-related efforts before an NRC decision on the suitability of the site has been made. Existing provisions of the NRC rules and regulations, Subpart F of 10 CFR 2 and Appendix Q of 10 CFR 50, do permit a review of site suitability issues prior to the filing of the CP application, but this review does "not constitute a commitment to issue a permit or license."

The industry argues that the non-binding nature of the existing site suitability provisions of the regulations discourages their use. Accordingly, DOE recommends that a regulatory framework be developed by NRC to approve sites or characteristics of sites independent of and prior to the filing of an application to construct a production and utilization facility on the site. The early site approval would be valid for a number of years and would

constitute an automatic Limited Work Authorization (LWA) at the time of application for a construction permit. In principle, sites could be banked by utilities, states, localities, or even possibly by developers.

The NRC now proposes to add a new section (193) to the Atomic Energy Act that authorizes the Commission to approve a site in the absence of an application for a CP or Combined Construction and Operating License (COL). The permit would be valid for 10 years, and could be renewed for not less than five nor more than 10 additional years. Utilities and Federal, regional, state, and local governments can apply for site approvals, thus encouraging broader planning. In contrast to the DOE proposal, a site permit does not constitute an "automatic LWA" upon the filing of a CP application for the site, probably because Section 193 specifically provides that the site may be used for other purposes than nuclear generation. These other uses may make the site less suitable for construction of a nuclear plant.

Early site approval proceedings could potentially involve the public at the earliest possible time, before substantial resource commitments have been made, and when public participation can be most effective. However, unless the banked site is eventually used for nuclear generation, the net effect on licensing time and resource expenditures would be negative. Not including the time required to bank a site, an estimate of six months (out of a total of 152 months) savings in lead time is attributed to early site approval by Los Alamos Scientific Laboratory (LASL).[12]

The Government Accounting Office (GAO) questions the time savings derived from early site approval,[13] citing preconstruction review statutes in 23 states, some of which determine the ultimate review schedule. In New York State, for example, the lengthy site review cannot be initiated until such time as a utility applies for a construction permit.

3.2 Preapproval of Designs

The multiplicity of nuclear power plant system and subsystem designs, and the variety of hardware used, dilute the base of experience necessary for the stabilization and maturation of the industry. Each custom design must be

tortuously dissected afresh in the licensing review, which is on the critical path for the project. The existing Appendices M, N, and O to Part 50 of the NRC Rules and Regulations allow for the review and approval of standardized plant designs, but the existing procedures do not render a binding approval. Indeed, it can be argued that as long as the regulations permit undisciplined backfit (see discussion below) of approved designs, incentives are absent for the industry to proceed in this direction.

The DOE proposal would allow for preapproval of entire facilities or facility subsystems independent of the application for a production and utilization facility license. The approval would be valid for a number of years, and would be stabilized by a strong backfitting provision. The NRC proposes to add a new Section 194 to the Act, essentially the same as the DOE proposal, with approvals valid for 10 years, renewable for not less than five nor more than 10 years.

Everyone is for standardization--the nuclear industry, the NRC, and the public. There are two issues, however. The first is the extent of standardization. The second is how to bring it about. They are related.

The aborted NRC proposal avoided the definition of "standardized plant designs" in the legislative language but defined it in the background information as "an essentially complete final design for a whole nuclear power plant usable at multiple sites." The new NRC draft proposal states that "typically, a final design should be described in such a manner that it could be used at most sites with a minimum of adaptations because of site-specific characteristics." However, both the NRC and DOE proposals allow preapproval of facility subsystems. Given the importance of systems interactions in plant safety, it is unclear whether certification of a portion of the plant would be practical. In a recent paper by TVA,[14] it is argued that "there is no such thing as a completely standard design unless the plant uses identical equipment throughout." In fact, the GAO argues that the review by the NRC in the late 1970s of two standardized plants took as long as custom plants because only the Nuclear Steam Supply System, not the Balance of Plant, was standard.[13]

In a recent assessment of standardization,[15] the Office of Technology Assessment (OTA) concludes that neither the industry nor the NRC will take the necessary steps voluntarily to adopt a standard "safety block" system or total plant design. This would require significant changes in our marketplace and government institutions.

3.3 Backfit

Backfit* is defined in 10 CFR Part 50.109 as the "addition, elimination, or modification of structures, systems, or components after the Construction Permit has been issued." Part 50.109 states that the Commission may order a backfit if it "finds that such action will provide substantial additional protection which is required for the public health and safety." Although this provision of the regulations has not always been explicitly cited, the history of nuclear reactor regulation is riddled with examples of identification of new safety issues, and the imposition of fixes on projects in the design, licensing, and construction stages. These fixes have infrequently been reviewed by a central authority or subjected to a rigorous cost/benefit analysis, giving the impression of an agency and a system run amuck.

DOE recommends that all backfits be subjected to centralized review and approval and that a backfit may be approved only if it is needed to "reduce the risk of the overall plant operation to an acceptable level." LASL estimates[12] that an effective backfit provision would shave 24 months (out of a total of 152 months) off of the lead time. AIF takes issue with the DOE language relating to an acceptable level of risk, preferring instead that "the value of the added protection provided by the backfit exceeds the increased costs."

The concept of "acceptable level of risk," strongly endorsed by the (NRC-sponsored) Rogovin Report[16] on Three Mile Island, is intimately related to the ongoing NRC Safety Goals[17] proceedings. Although the industry supports the concept of a Safety Goal, the AIF is opposed to the comprehensive implementation plan recently proposed by the NRC staff.[18]

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*The term "ratchet" is generally applied to a requirement on an operating plant.

With the formation of the Committee to Review Generic Requirements (CRGR), the NRC has already taken a big step toward the concept of a centralized management focus for the review, prioritization, and coordination of new requirements on licensees. The formation of this Committee under a Deputy Executive Director for Operations (EDO), with the backing of the Commission,[19] suggests a stronger role than the earlier Regulatory Requirements Review Committee (the "ratchet" committee). Moreover, to support the efforts of the CRGR, a cost branch is being formed to evaluate the economic impact of regulatory requirements.

The current NRC backfitting proposal[7] would amend 10 CFR Part 50.109 (and other sections for conformance), changing the definition of backfitting (to "the imposition of new regulatory requirements, or the modification of previous regulatory requirements applicable to the facility after the construction permit has been issued") and requiring that the change provide a "substantial" increase in plant safety and that the benefit exceed both direct and indirect costs. In addition, a new "backfit decision making process" is to be applied: the remaining life of the affected plants must be considered in determining costs and benefits; the NRC staff must analyze the effect of the backfit before it is required, unless the Commission finds immediate need for its application; licensees have an opportunity to demonstrate that a backfit is not needed in their case; and the EDO will review staff requests for information from licensees to assure that the request is valid. The proposed policy and rule would apply to backfits imposed by rule, regulation, or Commission order.

The Safety Goal issue is complex. The concept of a safety goal, like standardization, is universely popular. However, the concept is meaningless without an implementation plan, and this by necessity involves probabilistic risk analysis (PRA) to some degree. Intervenors as well as several members of the ACRS are cautious when it comes to PRA, maintaining that the methodology is not sufficiently developed and the results are too uncertain to use in the licensing process at present. The AIF may be expressing similar reservations in their objections to the implementation plan. However, it is not clear that the implementation of their (and NRC's) proposed alternative, balancing of the benefits against the costs, differs in any substantive way. One concept, it

would appear, involves the comparison of evaluated risks against a threshold and the other against evaluated costs.

3.4 One-Step Licensing

The existing two-step licensing process was developed shortly after the enabling legislation was adopted,[20] when nuclear reactor concepts were rapidly evolving and differed substantially from plant to plant. Because less than 25 percent of the design is usually completed by the time that the CP is issued, the project is vulnerable to backfit during construction as the design evolves. The existing process does not allow for meaningful public participation until substantial resources have been committed to the project, and the OL review may affect the construction process because of the diversion for regulatory review and hearings. Moreover, the concept of withholding an Operating License after literally billions of dollars have been invested in the project is so preposterous that the OL stage may be no more than a perfunctory exercise, thus reducing the credibility of the entire review. Both the Rogovin Report[16] and the President's TMI Commission[21] recommended abolishing the existing two-step licensing process.

In theory, a combined construction and operating license (COL) procedure would allow for detailed consideration of safety issues before the beginning of construction and before a significant commitment of resources has been made. The only issue is the level of detail required in the application. The level of detail in current FSARs cannot be provided because the detailed manufacturers' information is not yet available, while the level of detail in current PSARs is clearly not sufficient for the NRC to make a final determination that the plant is safe enough. And the determination would have to be final, with the backfit/stability provisions discussed earlier in effect, or the industry would not commit the necessary resources to apply for a COL.

For a custom plant, a one-step licensing procedure is estimated by LASL[12] to take longer, by approximately nine months (out of a total of 152 months) than the existing two-step procedure. For a pre-approved design and certified site, however, savings of 42 months are estimated. For this reason, and because a final design including manufacturers' specifications may be neces-

sary for one-step licensing, this proposal may only be practical for standardized plants. In fact, one-step licensing appears to make the most sense in conjunction with preapproved sites and designs, in which the licensing proceedings would merely be a "marriage" between the site and plant. NRC proposes[7] to take this approach wherever possible, though applicants may still request separate CP and OL licenses.

3.5 Hearings

The hearings' process represents the most emotionally charged issue in the controversy between the nuclear industry and the intervenors, and the issue with the greatest degree of polarization between the two sides. The existing process requires hearings at the CP stage, with optional hearings (generally requested) at the OL stage. Traditionally, although not required by statute, formal adjudicatory hearings are held. It is generally felt that this precedent is so ingrained that legislation would be required to change the system.

The two-step process frequently results in a replay of the CP contentions during the OL hearings. Moreover, licensing boards, in opening the proceedings to interested members of the general public, frequently permit contentions that are not solidly supported in a basis of fact. The format is conducive to considerable waste and irresponsibility. On the other hand, most of the negotiations and technical decisions are made informally between the applicant, the NRC staff, and their contractors well before the hearings, so that the intervenors feel that the deck is stacked from the start.

Both NRC and DOE propose a hybrid hearing process, which combines elements of both legislative and adjudicatory hearings, similar to the hearings' provisions of the National Nuclear Waste Policy Act of 1982.[22] Adjudicatory hearings could be granted by the hearing officer if there exists "a genuine and substantial dispute of facts that can only be resolved with sufficient accuracy by the introduction of evidence at a formal hearing." Moreover, the DOE proposal stipulates that an issue already settled in other procedures would not be allowed to be raised unless it can be shown that the attendant risk is increased beyond the "acceptable level of risk."

It is certainly not clear that trial-type hearings can effectively resolve technical issues. On the other hand, the intervenors maintain that informal hearings would effectively preclude intervenors and the public from participation in the licensing process. In any case, LASL estimates[12] that the DOE hearings reform would shave less than 3 months (out of a total of 152 months) off of the lead time.

The NRC has also proposed[7] a number of administrative reforms aimed at tightening up the hearings process. Briefly, they are:

- Strengthen the screening process
  - Establish an ASLAB group to act as a clearinghouse for all requests, contentions and petitions to intervene
  - Require that a potential intervenor establish standing
  - Raise the threshold for admissibility of contentions by requiring evidence that a real, factual dispute exists
- Improve the Conduct of Hearings
  - Minimize staff work and require stronger hearing panel control of the process
  - Rely on written material; the need for live testimony, including cross-examination, must be demonstrated
  - Permit the screening board or the hearing officer to appoint a panel of technical experts to determine the basis for conclusions and evaluating testimony
  - Limit sua sponte motions; permit summary motions
- Tighten the decision process
  - Reinstate the immediate effectiveness rule
  - Restrict intervenor participation to issues raised by that person

3.6 Mandatory ACRS Review

The enabling legislation requires the Advisory Committee on Reactor Safeguards to review each license application, at both the CP and OL stages, even if the Committee does not judge the review to be merited. The ACRS is comprised of 15 members, drawn largely from academia, who devote only part of their time to Committee activities. The Committee is particularly adept at revealing

previously unrecognized safety problems, but it has few resources to pursue them in depth.

Both the Rogovin Report[16] and the President's TMI Commission[21] recommended that the ACRS be relieved of its mandatory review responsibilities and be allowed to participate as a party in hearings. The NRC, in a 1977 review of the licensing process,[23] recommended that the ACRS be given discretionary authority on which applications merit its review, and the current DOE proposal suggests the same (with mandatory review reserved for standardized design approvals).

Although the current NRC proposal makes no mention of changing the status of ACRS review, this proposal appears to be non-controversial. The intervenors particularly welcome the participation by the ACRS in hearings.

3.7 Interface with States and Other Federal Agencies

There is considerable duplication of resources in several aspects of the environmental review between the individual states and the Federal government. Much of this is unavoidable, since the National Environmental Policy Act (NEPA) makes no provision for coordination with the states (23 of which have statutes requiring preconstruction review of power plants). GAO recommended that the NRC work jointly with all the states to identify common legal and procedural requirements as a first step in coordination.[13]

All parties appear to agree that the NRC is poorly qualified to judge the Need for Power, as required by NEPA. DOE proposes that an authorized state or local agency, or other public entity certify the Need for Power. The intervenors feel that the states are in the best position to certify the Need for Power. NRC proposes to accept the certification of any competent Federal, regional, or state governmental organization.

The Atomic Industrial Forum goes one step further in recommending that the NRC should cooperate to the fullest extent possible with other affected Federal agencies, states, or political subdivisions to assure efficient, timely, and non-duplicative reviews.[8] On the Federal level, NRC has developed agreements

with other agencies to expedite the environmental review. Elimination of duplication of state efforts, however, would appear to be a fertile ground for legislation.

3.8 Miscellaneous

Revise the "Ex Parte" Rule

The ex parte rule effectively cuts off communication between the Commissioners and other elements of the Agency during a licensing determination. The Rogovin Report[16] recommended more active involvement by the Commissioners in individual licensing determinations. NRC has proposed two rulemaking options to address the issues involved in the ex parte rule and separation of functions within the Commission during a license proceeding. The first option is to allow the Commissioners to talk to staff supervisory personnel who are not participating directly in a particular hearing; the second provides that the separation of functions rule does not apply in initial licensing decisions (the Administrative Procedure Act allows this exception). Linked with these issues is a proposal to limit the staff's participation in contested initial licensing proceedings to those issues on which it disagrees with the applicant's technical basis, rationale, or conclusions. These issues are by no means simple; arguments of constitutionality exist on both sides of the question.

In a related but much simpler proposal, NRC intends to bring the Commission more fully into licensing decisions by making the Atomic Safety and Licensing Appeal Board a Commission staff office that drafts Commission decisions, not an intermediate reviewer of ASLB decisions. Eliminating one layer of review should also help streamline the process.

Increase The Use of Rulemaking

The Rogovin Report[16] and the President's TMI Commission[21] recommended the increased use of rulemaking to resolve issues that affect several licensees. The NRC has been heading in this direction over the past decade: witness the ECCS, GESMO, Appendix I, and S-3 proceedings. However, considerably more

progress could be made in this direction. For example, the proposed NRC rule[7] allows generic factual issues that are resolved in a license proceeding to become proposed regulations within 45 days. If the Commission adopts the regulation following a public comment period, it cannot be relitigated unless "special circumstances," as defined in 10 CFR 2.758, are shown.

Eliminate CP Expiration

Section 185 of the enabling legislation provides for expiration of the Construction Permit if the facility is not completed within a designated period of time. Over the past several years, the frequency of applications for extensions has increased dramatically and these applications are perfunctorily approved. The provision currently serves only to generate paper and it is generally agreed that it should be eliminated.

Shed Some Responsibilities

The NRC is ultimately responsible for the anti-trust review, even though the opinion of the Justice Department is solicited and weighted heavily in the determination. Similarly, the NRC works closely with the Department of State in granting Export Licenses and in matters of non-proliferation. It has been asserted that these activities dilute the capability of the Agency to perform its primary function, assurance of the safety of nuclear power plants. A solution to this problem would be the transference of these functions in toto to the Departments of Justice and State, respectively. However, NRC's current proposal[7] does not suggest this.

Increase Public Participation

At present, significant public participation in the licensing process does not occur until the hearings. By then, however, most of the issues have been resolved through a series of informal meetings between the applicant and the NRC staff. The NRC proposed in 1977[23] to increase public participation during the staff review by holding meetings in the vicinity of the site and soliciting public involvement. This move would no doubt be popular with the intervenors. The current NRC proposal[7] requires notice that the Commission is

considering favorable action on an application to be published in national newspapers that are distributed in the vicinity of the site, as well as local newspapers.

Decentralize the Licensing Authority

Other countries, such as Germany, involve trade organizations directly in the regulatory process. In this country, the Institute of Nuclear Power Operations (INPO) was formed by the utilities after Three Mile Island to monitor power plant operations. Many of INPO's activities duplicate those performed by the NRC. Some cooperation between the two organizations has been initiated--in the standardization of incident reporting formats, for example. Shifting some of NRC's responsibilities onto INPO might be worthy of consideration.

Revise the Entire Fabric of the Regulations

The existing patchwork of NRC rules and regulations evolved over a 25-year period, each new rule devised on a largely ad hoc basis. The relative contribution of each of the numerous regulations to safety is largely unknown, although it is suspected to be highly non-uniform. It has been suggested that the NRC start from scratch, building its new set of regulations using probabilistic risk analysis as a tool, aiming toward quantitative safety goals.

Funding of Intervenors

The intervenors argue that intervention is necessary to assure that the NRC and the utility comply with all applicable standards, and that they are unable to do an effective job because of their lack of resources. This reduces their efforts to tactics which are viewed as wasteful and irresponsible by the industry. If the intervenors were funded, they could participate in the technical dialogue between the NRC and the applicant before the hearings, and the hearings could presumably concentrate on real technical issues.

Replace the Commission with an Administrator

Both the Rogovin Report[16] and the President's TMI Commission[21] recommended that the Commission be replaced with a single administrator. Since those recommendations were made, the Chairman has been given more power, two (Research and Standards) of the four staff offices have been consolidated, and the position of Deputy Executive Director for Regional Operations and Generic Requirements has been created in the EDO's office. Although these efforts have increased coordination and tightened up on management, the Agency still has a long way to go in the direction of efficiency. In keeping with the strengthened role of the EDO, licenses would be issued by that office, not NRR, under the NRC proposal.[7]

Consolidate at a Single Location

The Agency is spread out in at least six locations in the Washington metropolitan area. No other factor has contributed more to the lack of coordination and ineffiencies of operations. Consolidation has been discussed and attempted several times over the past several years, to no avail.

4.0 TECHNOLOGICAL OPTIONS

Commercial nuclear power plant licensing experience within the United States is, with only a few exceptions, based on the light water reactor (LWR) design concept [in particular, large (>1000 Mwe) LWRs]. The notable exceptions are the Fort St. Vrain high temperature gas-cooled reactor (HTGR), which is in operation, and the Clinch River Breeder Reactor Plant, a liquid metal (sodium) cooled reactor with an uncertain future.

Given the experience to date with the licensing of LWRs, it is reasonable to question whether a shift to small (<400 Mwe) LWRs and/or different reactor design concepts would have any significant impact on the existing licensing process. The technological options worth considering can be categorized as "safe" and "ultra-safe." For purposes of this discussion, it is assumed that licensed reactor designs are, by definition, safe. Recent Probabilistic Risk Assessments (PRAs) indicate that the probability of severe core damage for current LWR designs is on the order of 10^{-4} to 10^{-6} (per year).[24] Assuming, then, that a range of 10^{-4} to 10^{-6} constitutes a "safe" design, an "ultra-safe" design is one in which the probability of severe core damage is much lower than this.

There are currently several technological alternatives to the large LWR. These include, but are not necessarily limited to the following.

Small (<400 Mwe) LWR

Small LWRs contributed greatly to the original development of commercial nuclear power in the U.S. However, apparent economies of scale in capital costs motivated utilities to purchase larger reactors. There is a re-emerging interest in small nuclear power plants for reasons of both cost and safety, the latter resulting from the lower inventories of fission products available for release in the event of an accident.

A shift to smaller reactors could not be made by replicating existing small LWRs, since the designs of these plants do not meet all current safety requirements. In other words, existing small reactors are considered on the low side of our definition of "safe." The NRC has established a Systematic

Evaluation Program specifically to review these older designs and improve their safety where possible. A shift to smaller reactors would require new designs based on current NRC requirements. Except for the size of the core and the components in the plant, licensable small LWRs would not be expected to differ substantially in design from existing LWRs. Thus the licensing process for new small reactor designs would probably be quite similar to that for current large LWR designs, including the potential for backfit requirements.

Pressurized Heavy Water Reactors

Canadian (CANDU) reactors are almost exclusively of the pressurized heavy water design. There are certain aspects of the CANDU design which, in the opinion of some observers, make it a "very safe" (but probably not "ultra-safe") reactor. Compared to LWRs, there is little operating experience with the CANDU design. Moreover, Canadian reactor regulation is relatively lax in comparison with regulation in the U.S. Therefore, the licensability of the CANDU reactor in the U.S. would certainly be no better than that of the LWR, and most probably worse, considering the fact that the NRC would have to establish new design criteria and standard review plans to perform its review.

High Temperature Gas-Cooled Reactor

PRAs for the HTGR indicate that the probability of severe core damage may be lower than that of LWRs. However, operating experience with HTGRs is minimal. The difference in estimated risk between LWRs and HTGRs is probably not large enough to warrant the label of "ultra-safe" for HTGRs.

Factors that contribute to the safety of HTGRs include the large heat capacity of the graphite moderator, the simplicity of the shutdown mechanism (boron pellets in a hopper), and the fact that there can never be a total loss of coolant accident since the coolant is a gas. Safety considerations for HTGRs include the possibility of graphite fires, the very complicated design of the steam generators, and the fuel barrier integrity.

During the early 1970s, a number of utilities applied for construction permits for large HTGRs. As a result, the NRC made a significant effort to formalize design requirements and establish review plans for HTGRs. Thus the licensing framework for HTGRs is significantly better established than that for CANDU

reactors. Nevertheless, a number of years would be required to establish parity between the licensing framework for HTGRs and LWRs.

Other Reactor Concepts

A number of other reactor design concepts could be used in lieu of LWRs. Many of these were explored in the early development phase of nuclear power and include, for example, organic-cooled reactors, sodium-graphite reactors, and fluid-fueled reactors (molten salt). The risks inherent in these design concepts have not been assessed using current safety criteria. Accordingly, it is not clear whether any of these concepts are, by our definition, "safe," let alone "ultra-safe."

Ultra-Safe Reactors

Ultra-safe reactor designs are intrinsically safe; i.e., they do not require or depend on engineered safety features in order to mitigate the consequences of postulated events such as loss of electrical power or sabotage. An inherent advantage of ultra-safe reactor concepts from a licensing perspective is that they should not be susceptible to backfit requirements. Since these reactors are by definition orders of magnitude safer than current "safe" designs, it can be argued that costly backfits to make the plant safer could never be justified. In other words, if, after a thorough review a design can be shown to be ultra-safe, the margin of safety would be so great as to preclude any possibility of ever demonstrating the design to be unsafe. Uncertainties in the safety analysis would be well within the large safety margin, and the design could be licensed again without review.

Are there any ultra-safe reactor designs in existence? A report prepared for the Institute for Energy Analysis[25] describes a Swedish concept called PIUS (Process Inherent Ultimate Safety), which is alleged to be ultra-safe. This interesting conceptual design has received favorable reviews.[24] However, the reactor at present is merely a paper conceptual design. Much work remains to be done before the PIUS design can be confirmed to be truly ultra-safe, let alone commercially viable.

5.0 DISCUSSION

Where is the major bottleneck in getting plants licensed and operating? Certainly the CP proceedings could be shortened by instituting more efficient staff review and reform of the hearings process. But here we are talking about months, rather than years. Moreover, investment in the plant is relatively small during this time. The most recently completed CP proceedings, for Yellow Creek, took 28 months from the date of docketing. Thus the CP licensing process is not the most significant problem.

Nor is the OL process the most significant problem. The OL proceedings are not on the critical path because they are conducted concurrently with construction. There was concern in the 1980-81 time frame that because of the diversion of NRC resources by TMI, a number of OLs would be delayed beyond the completion of plant construction.[26] Accordingly, an interim licensing bill was passed by the 97th Congress[27] to grant low power (5 percent of full power) operating authority prior to NRC issuance of the OL. However, none of the nine plants originally affected has yet been delayed by licensing proceedings, and only one may ultimately require interim licensing authority.

The real issue is the stability and extent of regulatory requirements. As the NRC Executive Director for Operations points out, there are 87 ways that regulatory requirements can be imposed.[28] There is evidence beyond a reasonable doubt, from our limited case studies as well as numerous other sources, that new and/or changing regulatory requirements and standards have significantly lengthened construction time over the past decade, in some cases by several years. We found that operating experience at other reactors (i.e., Browns Ferry, TMI, BWR Mark I containment) accounted for a significant fraction of the new regulatory requirements, if not the schedule slippage. Field construction problems stemming from what may be ill-conceived seismic criteria can be directly blamed for a sizable fraction of the schedule slippage during the last few years. Were these criteria necessary and were they interpreted appropriately?

But as others have pointed out, regulations and standards are not the only reason for lengthened construction periods. We found at least one instance of a deliberate slowdown because the plant was simply not needed at the originally scheduled date of completion. And our case studies also contain examples of management and coordination problems.* Several of the utility and A/E staffs were doubling in size each year in the mid-1970s. Moreover, decision-making chains had become more complex for all parties.** Other reasons for schedule slippage surfaced, but are probably of second order. A quantitative assessment of the contribution of regulatory requirements to delay is beyond the scope of this paper, and may not be possible in any case.

A cursory examination of other reactor types suggests that the selection of a design with longer times between loss of coolant and major fission product release, or lower fission product inventories than current large LWRs, may not result in a significantly shortened licensing process. Moreover, with the exception of smaller or redesigned LWRs, no commercial fuel cycle exists in the U.S. for these other designs. The ASEA-ATOM PIUS design does not even have the benefit of an operating prototype plant. If a different reactor type were to replace the present LWR, it would take a number of years to put into place another regulatory regime.

The following paragraphs address some of the proposals for regulatory reform. In order to provoke discussion, we have organized the material under four topics:

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*One of our case studies demonstrates the converse, an example of exemplary management performance. When a scheduling problem first surfaced, the utility project manager and NRC management sat down together and worked out an innovative solution.

**In the late 1960s, when a decision was required of the NSSS vendor, it could be rendered quickly. In the mid-1970s, the impact on several other customers with the same NSSS design had to be considered.

- Administrative Change within the NRC
- Regulatory Stability
- Acceptance of Design
- Public Involvement

5.1 Administrative Change Within the NRC

In evaluating the need for change, it is important to view the NRC as it is today and not as it was several years ago. This is difficult, since the call for change results from data obtained over a 10-15 year period. The regulatory environment today is different from that of the mid-1970s, or even early 1980, one year after TMI. Much of the recent change came about as the result of a survey of utilities by senior NRC management, in which it was alleged that the flurry of post-TMI requirements acted to diminish, rather than enhance, safety.[29] Whether this is true or not, there is a significant movement toward reform within the NRC.

The Committee to Review Generic Requirements, under a newly created position of Deputy Executive Director for Operations (EDO), was formed to provide a centralized management focus to review, coordinate, and prioritize requirements, as called for in the legislative recommendations by DOE. A cost branch is being formed to provide data to the EDO for the analysis of the cost-effectiveness of new requirements. A regulatory reform package has been prepared.[7] Safety goals have been published for comment,[17] and a draft implementation plan submitted to the Commission for review.[18] The Commission has circulated to the staff a Policy and Planning Guidance Document,[19] in which the following guidance was provided to improve the regulatory process:

- Eliminate all unwarranted delay in reaching regulatory decisions
- Evaluate requirements on a cost-benefit basis
- Address as many issues as possible in the context of a rulemaking rather than case-by-case review.

The Rogovin report[16] called the reactor licensing process "a sham." The Commission has moved toward accommodating the recommendations contained in this as well as the Kemeny Commission's[21] review of the NRC. The Offices of Standards and Research have been consolidated, an Office for Analysis and

Evaluation of Operational Data has been established, an Office of Human Factors Safety has been formed, and rulemaking (as opposed to case-by-case review) is being emphasized. The Commission, however, has not been replaced by an Administrator.

It can be argued that several of the recommended regulatory reforms are either already in place or can be accomplished administratively without the need for legislation. As discussed above, moves toward centralized review and a safety goal, key features of the DOE backfit proposals, have been initiated. Provisions of the regulations already exist that allow NRC to review site suitability and approve standardized plant designs. These decisions, however, are currently made outside of the process for considering an application for a license to construct a production and utilization facility, and are thus not binding on these considerations. The issue, then, boils down to stability.

Even the form of the hearing process could, in theory, be changed administratively. However, it is argued, and probably rightly, that the tradition of adjudicatory hearings is so ingrained that legislation would be required to shift over to a less informal process. NRC has proposed this change.[7]

The Chairman of the Atomic Safety and Licensing Appeal Board feels that legislative reform of the hearings process is uncalled for.[30] Rather, less drastic changes to improve the hearings process could be accomplished administratively. Proposed changes include:

- Pre-hearing filing of cross-examination plans to allow licensing boards to eliminate duplication.
- A more stringent test for admissibility of contentions
- Handling more issues on a generic basis

Optional ACRS review, transfer of certain non-safety functions to other agencies, and limitation of the NEPA review would indeed require legislative action. However, Commissioner Gilinsky argues[10] that even one-step licensing could be accomplished by changing the regulations to require applicants to submit an essentially complete design as part of the CP application.

In view of all of the foregoing, is legislative reform really necessary?

5.2 Regulatory Stability

As discussed at the beginning of the previous section, it is important to view the NRC and the regulatory process as it is today, and not as it was in the previous decade during the flurry of licensing activity, or immediately after the trauma of TMI. Indeed, a strong argument could be made that the process has already stabilized. The flood of industry standards and regulatory guides appears to have subsided. Only three I&E Bulletins have been issued to date in 1982. A well-documented, debugged Standardized Review Plan is in place. Hundreds of plant-years of operating experience have been amassed with large LWRs and mechanisms for assessing operating experience have been put into place. Literally hundreds of thousands of man-hours have been allocated to hunting for common mode failures, systems interactions, and overlooked accident initiators. TMI may have served as a final venting of the system with respect to "safety skeletons in the closet."

A useful way to look at the process is through the concept of "technological time constant." In order to get the bugs out of a new technology, experience must be gained over complete cycles of design, construction, and operation. In most engineering systems, these cycles are on the order of months or at the most years. For reactor systems, this technological time constant is on the order of a decade. We may have completed two time constants in the case of reactor technology. Is this enough?*

An argument against the perception of stabilization in the licensing process is the large number of outstanding safety issues -- approximately 160 in 1981. However, the staff has established an initial system for prioritizing these issues, and, as strongly recommended by the Kemeny Commission, has developed a plan for their early resolution.[31] The degree of staff success in accomplish-

*It would be interesting to pursue this concept through analogy with other technologies (large boilers, dams, transportation systems, etc.).

ing this plan may well answer the question about the stabilization of nuclear reactor regulation.

Another event that has reinforced the industry's perception of continuing chaos in the licensing process is the Sholly Decision. The NRC has routinely approved license amendments (roughly 400 requests per year) without holding public hearings if it determined that the amendment involved no significant hazards consideration. In the Sholly Decision, the U.S. Court of Appeals for the District of Columbia (November 19, 1980) ruled that an interested person who requests a hearing is entitled to such a hearing, regardless of the NRC determination on significant hazard. The NRC appealed, and the Supreme Court agreed to hear the case.

The use of Safety Goals or a cost-effectiveness prescription to stabilize the regulatory process carries attendant risks. If the existing regulatory process, based on technological standards, has indeed stabilized, the change to a performance standard could move the licensing process in a totally new direction, effectively destabilizing the system. Indeed, the draft implementation plan,[18] which relied heavily on Probabilistic Risk Analysis (PRA), gave hints of such a radical departure.

Underlying the existing regulatory process is the "defense-in-depth" concept for nuclear plant design. It can be argued that the reality of a shift from defense-in-depth to a safety goal concept entails only a change from event tree to fault tree formalism. Both formalisms suffer from the weakness that a foreknowledge of the spectrum of possible failures is required.

PRA is a useful tool for systematically examining systems or groups of systems in an integrated fashion to compare alternative designs or improvements. However, the bottom-line results of PRAs, particularly in terms of accident consequences, are so highly uncertain as to possibly negate their usefulness.[32] Because of this, the Safety Goal draft implementation plan focused on the calculation of a plant performance numerical guideline, the probability of a large-scale core melt. A bill currently before Congress[33] would establish a program under the coordination of the Office of Science and Technology Policy to improve the use of risk analysis by Federal agencies,

including the NRC. Is it premature or even practical to rely on PRA for licensing?

If it should turn out to be impractical to establish a systematic analytical method (such as the Safety Goals) to determine the desirability of backfit or new regulatory requirements, the decision could be made on the basis of judgment by a third party. The Committee for the Review of Generic Requirements has already been established under the Deputy Executive Director for Operations to review, prioritize, and coordinate requirements on utilities. It has also been suggested that the ACRS could fill this role, should its responsibilities for review of each license application be lifted. In either case, it remains to be seen if this third party will be able to gain control over the numerous mechanisms available to the Agency for imposing requirements.

5.3 Acceptance of Design

The issue in one-step licensing involves the finality of the design. There is little question that single stage licensing would be desirable. It would eliminate the charade of the OL stage, bring the public into the process at the earliest possible time, and remove the distraction of staff review and hearings during construction. However, to what extent would the design have to be "set in concrete" and how much of the actual hardware would have to be specified before the staff would be in a position to sign off on acceptability? NRC has stated that "an essentially complete final design" would be required.[7] The one-step review itself, for a custom plant, would probably take longer than the current CP review.[12]

With all of these considerations in mind, it would appear that one-step licensing would be most practical for a preapproved complete design, and for that matter, as a mechanism for linking this with a pre-approved site. Indeed, according to the LASL analysis,[12] herein lies the real potential for significant savings in lead time. The logic can be carried one step further. If an applicant were to go to the trouble to certify a complete plant design, including balance-of-plant, it seems likely that he would plan to use this pre-approved plant at a number of sites.

All of this begins to sound a great deal like a standardized plant. Standardization has been proposed as a cure-all for the ills of the nuclear industry for at least a decade. Indeed, Appendices M, N, and O of 10 CFR Part 50 were adopted in the 1974-75 time frame. All of the major NSSS vendors have submitted standardized designs to the NRC for approval. Still, every operating plant and every plant under construction is unique.

What will it take to achieve standardization in the nuclear industry? The first step is to define it. Pre-approved systems and subsystems, as described in the DOE licensing reform recommendations, would not appear to be sufficient. In fact, most knowledgeable critics would argue that a standardized plant that could accomplish the desired degree of industry stabilization would have to be standard down to the last nut and bolt. Some feel that the plant would have to be carried all the way through licensing and operation before it would qualify.[14] But is it actually feasible to standardize hardware? Is it possible to order the exact same equipment used in a previously constructed plant? How much of the plant can be standardized, considering the variability in balance-of-plant requirements for different sites?

Other countries have come a long way toward the ideal of standardization. The French have one utility, one constructor of NSSSs, and one constructor of turbine-generators. The Germans train and certify all of their operators in one powerplant school. In the U.S., there are 65 nuclear utilities, 14 nuclear A/Es, and four NSSS suppliers; there are 28 existing A/E-NSSS combinations.

A recent OTA study[15] concluded that standardization, although no panacea, holds promise of time and cost reductions and safety gains. Single-stage licensing and the adoption of a national safety goal were suggested as means to encourage standardization. (The original NRC legislative proposal[10] was a step in this direction.) However, the study concluded that standardization would also bring about significant and perhaps disruptive changes in the institutions of the commercial nuclear industry. Moreover, it is probable that neither industry nor the NRC will take the necessary steps voluntarily; Congressional action will probably be needed.

Should revolutionary changes in institutions not be desirable or possible at this time, less drastic changes could come a long way toward improving the licensing process. One proposal is to institute a system based on design criteria followed by independent verification. The existing licensing process involves an exhaustive review by the NRC staff of the detailed plant design. This resource-intensive process could potentially be supplanted by a system in which the licensee commits to specific design criteria, followed by independent verification by a third party. In theory, the CP could be issued after the staff is satisfied that the commitments are appropriate, and an OL could be granted following the verification that the plant is designed and constructed in accordance with the criteria. Case Study #2 illustrates a pilot study involving an Independent Design Review process.

5.4 Public Involvement

Discussion of the hearings process evokes the strongest emotions on all sides of the nuclear power controversy. Technologists are frustrated by the socio-political process played out at the hearings, a process that they do not understand and which is certainly not efficient for settling technical issues. The intervenors are upset by a deck that they perceive to be stacked against them. Nobody is particularly satisfied with the existing system. Yet the intervenors are loathe to see the traditional adjudicatory format altered, fearful that they would be cut out of the licensing process altogether. Do the intervenors speak for the public?

For all of the industry's insistence on the waste inherent in the existing hearings format, LASL estimates[12] less than three months savings in lead time from the proposed DOE hearings reform. Surely this three month investment, less than two percent of the current average lead time, is well spent if it permits public involvement and results in increased public confidence in the technology. However, is effective public participation served by the current hearings process? Industry proponents maintain that the hearings serve only as a bandstand for those intent on foreclosing the nuclear option.

Earlier and more effective public involvement is clearly in the best interest of all of the participants in the licensing process. Theoretically, the

public can get involved after the first pre-hearing conference. In practice, however, public involvement does not occur until the actual hearings. The NRC, in the 1977 evaluation of potential licensing improvements,[23] recommended conducting early informal meetings in the vicinity of the site. Early site approval proceedings could accomplish the same thing, at least with respect to environmental issues.

Without question, public confidence is necessary if there is to be a future for nuclear power.[34] To achieve public confidence, accident-free operation of existing plants must be accrued. Additionally, the public, including the partisans, must be provided an effective means to participate in the licensing process. Until a format acceptable to both sides is developed as a replacement, the inefficiencies in the current formal hearings may be a small price to pay to assure a measure of public respect for the process. Moreover, until such time as the political issue surrounding the acceptability of nuclear power is resolved, an efficient hearings format may not be achievable.

ACRS participation in the hearings is considered by representatives of both the industry and the intervenors as a positive step in the direction of reform. The current NRC proposal of using an "expert panel" to resolve technical issues in the hearings reflects this.

There is clearly room for more coordination between the NRC and the states in the environmental review of license applications. Some attempts have been made at joint NRC/State hearings, but the differing rules have made this difficult. Only legislation could shift some of the Federally-mandated NEPA responsibilities, such as the certification of Need for Power, to the states. This issue goes far beyond Need for Power, or for that matter, NRC's responsibilities under NEPA. The issue of state/Federal interactions under NEPA could potentially be considered by the Executive Council on Environmental Quality (CEQ).

6.0 REFERENCES

1. Report of DOE Task Force on Nuclear Licensing and Regulatory Reform, supporting paper to DOE press release, "Energy Department Considers New Licensing Reform Proposals for Nuclear Power Plant Construction," October 15, 1982.

2. Edison Electric Institute, 1981 Year End Electric Power Survey.

3. The Need for Regulatory Reform, United Engineers and Constructors, September 1981.

4. Regulatory Reform Case Studies, United Engineers and Constructors Report UE&C 820830, August 1982.

5. William E. Mooz, Cost Analysis of Light Water Reactor Power Plants, Rand Corporation Report R-2304-DOE, 1978.

6. Letter from Karl Cohen to Nunzio J. Palladino, Chairman, U.S. NRC, May 21, 1982, incorporated in file of correspondence on the proposed NRC Safety Goal, NUREG-0880.

7. Draft Report of the Regulatory Reform Task Force, U.S. Nuclear Regulatory Commission memorandum SECY-82-447, November 3, 1982, James Tourtellotte, contact.

8. Letter from Carl Walske, President of the Atomic Industrial Forum, to Mr. Bernard C. Rusche, Special Assistant to the Secretary of Energy, October 9, 1981.

9. Letter from Carl Walske, President of the Atomic Industrial Forum, to Mr. Bernard C. Rushe, Special Assistant to the Secretary of Energy, January 15, 1982.

10. Nuclear Standardization Act of 1982, proposed legislation, May 27, 1982.

11. Memorandum from James R. Tourtellotte, Chairman Regulatory Reform Task Force, to NRC Commissioners, U.S. Nuclear Regulatory Commission memorandum SECY-82-399A, October 6, 1982.

12. R. H. Drake et al., "Quantitative Analysis of Nuclear Power Plant Licensing Reform," Los Alamos National Laboratory Report LA-9519-MS, September 1982.

13. "Reducing Nuclear Powerplant Leadtimes: Many Obstacles Remain," Report to the Congress by the Comptroller General of the United States, EMD-77-15, March 2, 1977.

14. "Revitalizing Nuclear Power Plant Design and Construction: Lessons Learned by TVA," presented to the AIF Workshop on Reactor Construction and Operation, October 1981.

15. "Nuclear Powerplant Standardization: Light Water Reactors," Congress of the United States Office of Technology Assessment Report OTA-E-134, April 1981.

16. "Three Mile Island, A Report to the Commissioners and to the Public," Mitchell Rogovin, Director, Nuclear Regulatory Commission Special Inquiry Group, January 1980.

17. "Safety Goals for Nuclear Power Plants: A Discussion Paper", U.S. NRC Report NUREG-0880 (For Comment), February 1982.

18. Memorandum from William J. Dircks, NRC Executive Director for Operations, to the NRC Commissioners, "Action Plan for Implementing the Commission's Proposed Safety Goals," July 6, 1982.

19. U.S. Nuclear Regulatory Commission Policy and Planning Guidance 1982, U.S. NRC Report NUREG-0885 Issue 1, January 1982.

20. The Atomic Energy Act of 1954 As Amended, Public Law 83-703 (68 Stat. 919).

21. "Report of the President's Commission on the Accident at Three Mile Island," John G. Kemeny, Chairman, Washington, D.C., October 1979.

22. National Nuclear Waste Policy Act of 1982, S.1662, April 29, 1982.

23. "Nuclear Power Plant Licensing: Opportunities for Improvement," U.S. NRC Report NUREG-0292, June 1977.

24. A. Fraas, Survey and Assessment of the Technological Options Available to the Nuclear Industry in the 1980 to 2000 Period, draft report, September 20, 1982.

25. K. Hannerz, Toward Intrinsically Safe Light Water Reactors, Institute for Energy Analysis draft report, June 14, 1982.

26. Licensing Speedup, Safety Delay: NRC Oversight, Ninth Report by the Committee on Government Operations, House Report No. 97-277, October 20, 1981.

27. H.R. 2330 and S.1207, 97th Congress, "Temporary Operating Licensing Authority."

28. Regulatory Reform - It May, or May Not, Solve Project Delays, Nuclear Industry, July 1982.

29. A Survey by NRC Management to Obtain Viewpoints on the Safety Impact of Regulatory Activities from Representative Utilities Operating and Constructing Nuclear Power Plants, U.S. NRC, July 1, 1981.

30. ASLAB Chairman Lists New Rules He Says Would Help Licensing Process, Inside N.R.C., October 18, 1982.

31. Memorandum from William J. Dircks, Executive Director for Operations, for the Commissioners, "Plan for Early Resolution of Safety Issues," U.S. Nuclear Regulatory Commission memorandum SECY-81-513, August 25, 1981.

32. ACRS Decides One Good Debate on Safety Goals Demands Another, Inside NRC, August 23, 1982.

33. H.R.6159, 97th Congress, "Risk Analysis Research and Demonstration Act of 1982," August 3, 1982.

34. Remarks by Nunzio J. Palladino, Chairman U.S. NRC, at the American Society for Quality Control, Ninth Annual National Energy Division Conference, Orlando, Florida, October 13, 1982.

CASE STUDIES ON REGULATORY REQUIREMENTS

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September 1983

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ENVIRONMENTAL QUALIFICATIONS OF ELECTRICAL EQUIPMENT

1.0 INTRODUCTION

1.1 Purpose

The purpose of this paper is to describe the Nuclear Regulatory Commission's (NRC) requirements on assuring electrical equipment used to mitigate accidents at nuclear power plants are capable of withstanding the consequences of the accident. This assurance process is generally called the Program for Environmental Qualifications of Electrical Equipment (or EQ program).

1.2 The EQ Program

Essentially, the EQ program provides a series of requirements, recommendations and guidance on the way in which utilities that own and operate nuclear power plants will qualify equipment for use in the plant. For example, a pump used to deliver water to the core of a reactor during an accident needs to be able to withstand the elevated temperatures, humidity and radiation levels created by the accident, say a broken pipe. Therefore, test analysis or combination of tests and analysis should be done to the pump's motor under conditions which simulate the environment created by the accident to make sure the pump will run when the pipe breaks and water is needed to cool the core. This same program of qualification applies to the rest of the electrical equipment in a nuclear power plant such as valve motors, instruments, sensors, cabling, meters, etc.

2.0 HISTORY

NRC bulletins on the qualification of electrical equipment date back to November 8, 1977. Inspection and Enforcement (IE) Bulletin 77-05 required action by licensees on pin and socket type electrical connectors that had failed tests at Sandia Laboratories. These tests showed that many of the connectors failed to survive simulated LOCA environments. The bulletin, therefore, only addressed to those connectors inside the containment. The type of failures seen were split shrink tubing on pin cables and no adhesion of potting material to the connector shell. Responses to this bulletin were required in 30 days for Operating Licenses (OLs) and 60 days for Construction Permits (CPs).

One week (November 15, 1977) after 77-05 was issued, Supplement A to the Bulletin was sent to the same set of licensees. This expanded the scope of 77-05 "to include all connectors in safety systems which are required to function to mitigate an accident..." Again, OL's were to respond in 30 days and CP's in 60 days.

On November 22, 1977, about one week after Supplement A to 77-05 was issued, IE issued Bulletin 77-06. This bulletin described the shorting of conductors in an electrical connector inside the containment of Millstone Unit 2 which caused a normally closed valve in the safety injection system to open. Additional resistance testing of other connectors revealed connectors with reduced insulation resistances. The cause of the shorts were moisture accumulation and small cracks in the epoxy of the containment penetration assembly. The Bulletin asked that licensees review their containment penetration assemblies and report what measures will be taken to assure their integrity. The response was due in ten days.

About one year later, in early 1978, IE issued Bulletin 78-02 describing tests performed by the Franklin Institute on unprotected terminal

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blocks. The Bulletin does not describe the exact cause of the failure. However, it required licensees to determine if the type of terminal block that failed was used in their plants in areas subject to the accident environment and what action they would take to assure they were environmentally qualified. OL's were required to respond in 14 days and CP's in 30 days.

About two weeks later, IE issued Bulletin 78-04 which described the use of unqualified, stem-mounted limit switches installed in motor and air operated valves in the containment. Attached to the Bulletin was a Westinghouse Technical Bulletin describing two alternatives for correcting the problem. The IE Bulletin required licensees to determine if such limit switches were used in their plants in safety-related equipment in the primary containment and what action they would take to assure qualification. OL's had to report in 30 days and CP's in 60 days.

In May of 1978, about three months after 78-04 was issued, IE issued Circular 78-08. This circular was in response to a Commission order in April which responded to a Union of Concerned Scientists (UCS) petition. The purpose of the Circular was to bring to licensees' attention deficiencies found in the responses to the previously-issued bulletins and to "highlight the important lessons learned". The circular recommends that licensees "should examine installed safety-related electrical equipment, and ensure appropriate documentation of its qualification to function under accident conditions." An answer to the circular was not required and there was no due date for completion of the review.

In February of 1979, IE issued Bulletin 79-01. This Bulletin escalated the review recommended in Circular 78-08 to a requirement. This escalation resulted from NRC inspections of licensees' actions in response to 78-08. The bulletin stated that the "resolution of problem

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areas are not receiving the level of attention from all licensees which the NRC believes is warranted." The Bulletin required the reviews required in 78-08 be completed in 120 days.

At this point, the history of environmental qualification of electrical equipment becomes complicated. Bulletin 79-01 has Supplements A and B. Supplement B has three additional supplements. The Bulletin and Supplements stretch out over about a year and a half. Supplement A deals with a part defect report (Title 10 Code of Federal Regulation part 21 on ASCO solenoid valves. The requirements in this supplement were tied to the due date in 79-01. Supplement B requests more detailed information than was required in 79-01 and increases the scope of the Bulletin to include high energy line breaks inside and outside the containment, aging and submergence. It also includes the NRC's Division of Operating Reactor (DOR) Guidelines which the staff were to use to evaluate the licensees' submittals.

In March of 1979, about a month after 79-01 was issued, the Three Mile Island (TMI) accident occurred. Utilities now had to purchase additional equipment required by the TMI Action Plan. This equipment included the subcooling monitor, meters to read the exit core thermocouples, containment high radiation monitors, water level indicators, etc. The staff and the utilities were faced with deciding whether or not this equipment was to be qualified according to the requirements in 79-01B.

In May of 1980, the Commission issued its Memorandum and Order (CLI 80-21). The Commission ordered that the requirements in the DOR Guidelines "form the requirements which licensees and applicants must meet in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criteria (GDC) 4 which relate to environmental qualification of safety-related equipment." In October of 1980, technical specifications changes were sent to operating reactors making the above a firm requirement subject to enforcement. The changes were explicit. The

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first required that "all safety-related electrical equipment shall be qualified in accordance with the provision of the DOR Guidelines or NUREG-0588". The second change required "complete and auditable files be available and maintained at a central location which document the degree of compliance with the DOR Guidelines or NUREG-0588." The due date for qualification was June 30, 1982 and for the files was December 1, 1980. Two generic letters (81-05 and 82-09) were sent after the technical specification changes were issued explaining and clarifying the staff's position on replacement parts, cold shutdown, TMI-related equipment, the June 1982 deadline, mild environments, etc.

In January of 1982, the proposed rule on EQ was published in the Federal Register. The qualification deadline was extended to the second refueling outage after March 31, 1982. In May of 1982, the staff sent the Commission the final rule on EQ. Public comments were focused on ten areas, five of which were difficult for the staff to resolve. A month later, following a Commission meeting with the staff, two staff requirement memos modifying the rule and requiring a status report form the staff were written. Later in the month, the Commission voted to suspend the deadline for EQ because a decision on the final form of the rule containing the deadline extension could not be reached and the deadline in the Memorandum and Order (80-21) was a week away. In the meantime, two versions of the final rule were written in response to the questions from the Commissioners. CRGR then reviewed the rule and recommended to the EDO that the Commission approve it. The Commission approved the final rule in December of 1982 and it was published in the Federal Register on February 22, 1983 (48 FR 2729).

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3.0 EQUIPMENT COVERED BY THE EQ PROGRAM

3.1 Safety-Related Equipment

General Design Criteria (GDC) 4 requires equipment used to mitigate an accident be qualified to operate in the environment created by the accident. The accidents are those analyzed in the plant's FSAR. For the low pressure injection system, this means that those portions of the system inside the containment be capable of withstanding the heat, humidity and radiation exposure of the containment after the largest pipe on the vessel falls off. For electrical equipment, this includes motor operators on valves, the instrumentation used to generate the injection signal, the containment electrical penetration and connecting cables. For equipment outside the containment, such as the pumps and motor control centers in the auxiliary building; GDC 4 would apply in the same manner. For example, if no credit in the analysis is taken for off-site power during the LOCA, then the equipment must operate continuously in the auxiliary building without the normal building ventilation system. If a steam line runs through the room in the auxiliary building where the equipment is located, then the equipment must survive the line breaking if the equipment next to the line is used to cool the core when the pipe breaks. This would include the auxiliary feedwater pumps for PWR's and the high pressure injection systems in BWR's.

3.2 Equipment Important to Safety

The NRC's experience, however, shows that the analysis in an FSAR does not always represent what happens in a real plant. The Browns Ferry fire showed that condensate pumps in the feed and condensate system which are not safety-related are important in keeping the core cool. The Ginna tube rupture showed that the PORV, which is also not safety-related, is important in limiting the amount of radioactive liquid dumped into the

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environment. In addition, the staff did not consider those events terminated until the plant was in cold shutdown. Therefore, it seems reasonable that the staff throught the list of equipment requiring environmental qualification should be extended to equipment to complete one path to cold shutdown. Appendix A to the DOR Guidelines, which is repeated in supplement 2 to 79-01B, lists equipment to be qualified and includes equipment needed to achieve cold shutdown. However, this list is mixed up with design basis accident equipment and the requirement is not clear. The staff backs off (clarifies??) its position on cold shutdown in Generic letter 81-05 (January 19, 1981) by telling utilities that they need only to qualify equipment to hot safe shutdown if that was their licensing basis. In the same letter, however, the staff sticks by the 79-01 requirement that utilities should provide the staff a list of equipment to achieve cold shutdown and whether or not it is qualified. Finally, the staff told several of the Commissioner's technical assistants in a meeting in September of 1982 that this requirement for a list was a "gimme" and the staff has not and probably would not review the information requested in the letter.

3.3 Instrumentation to Follow the Course of an Accident

The scope of the equipment to be covered by the EQ program may include TMI Action Plan equipment and the instruments used by the operator in the plant's emergency procedures. The staff tells the utilities in Supplement 1 to 79-01B (February 29, 1980) in the answer to question 5 that equipment modified as a result of actions, such as "Lessons Learned", need not be reported in their response to 79-01B. Later, in Supplement 2 to 79-01B (September 30, 1980) in the answer to question 21, the staff says the TMI Lessons Learned instrumentation is subject to the same requirements as other safety-related equipment. In Supplement 3 (October 24, 1980) to the Bulletin, the staff restates and clarifies this position (see item 2). In generic letter 81-05 to licensees, the staff

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tells licensees to refer to Appendix B of NUREG-0737 (Clarification of the TMI Action Plan) for information on the qualification of TMI-related equipment. In generic letter 82-09, the staff tells utilities that all display information used in the emergency procedures need not be identified. The staff goes on to say that if the utility has already submitted the list of TMI-related equipment, the NRC will not review the information until other activities are implemented. In Supplement 1 to NUREG-0737 (September 8, 1982) (see SECY 82-111B), the staff told utilities that the SPDS need not be designed to the requirements of GDC 4. The same statement applies to the instrumentation in the Technical Support Center and the Emergency Operation Facility.

3.4 Replacement Parts

The first mention of spare parts is in the first Supplement to 79-01B (February 29,1980). The staff told utilities that spare parts are to "meet the same criteria as the installed electrical equipment or component resulting from the evaluation of IEB 79-01B." This position is restated in Supplement 2 to the same bulletin with the addition that "parts used to replace presently-installed parts shall be qualified (to the latest standard (IEEE-323 1974)) unless there are sound reasons to the contrary". In generic letter 81-05 (January 19, 1981), the staff says it is the licensee's responsibility to justify deviation from the latest standard in an auditable manner. "Sound reason to the contrary" are a matter of degree and will be judged accordingly. In generic letter 82-09 (April 20, 1982), six sound reasons to the contrary are listed. In SECY 82-207D (August 26, 1982), the Commission is given three options on the use of "in-kind" replacement parts: 1) indefinitely, 2) through November 30, 1985, 3) through another date.

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4.0 DISCUSSION

As can be seen from the above, many of the requirements on EQ for electrical equipment are ambiguous and inconsistent. The NRC staff was probably correct in saying that the level of attention that utilities had given environmental qualification prior to tHe Commission's Order (80-21) was inadequate. The purpose of the Order was to make clear that non-compliance with the DOR Guidelines and NUREG-0588 would be the basis for enforcement action. However, the state of compliance with the order is apparently not known. On July 6, the Commission asked for a status report by the end of August on the evaluations of licensee submittals on EQ. This status report should have shown whether or not the provisions of the proposed rule were reasonable. It would also show whether or not he utilities had given the problem adequate attention. The staff did not respond to the request. Therefore, the Commission did not know at the time of the decision on the final EQ rule was required (July 1982) if their requirements were reasonable and were taken seriously.

Sorting out the requirements in the final rule is difficult. If the NRC's traditional philosophy of regulating were followed, there would be no need for the rule. GDC 4 must be followed. The staff's guidance for compliance with GDC 4 is given in the DOR Guidelines and NUREG 0588. Operating reactors licensed prior to Sequoyah must comply with these qualification requirements because they were made part of their operating license. Plants licensed after Sequoyah will comply with the latest standard (IEEE-323 1974) in the Standard Review Plan which includes the requirement of the DOR Guidelines and NUREG-0588. If owners of these plants decide to comply with GDC 4 by other methods than the latest standard, the staff must review and approve the methods prior to issuing an operating license. The net effect of implementing the final version of the rule on existing qualification programs for electrical equipment was zero.

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5.0 CONCLUSION

It is difficult to draw definitive conclusions from this example of the rulemaking process. In general, however, the degree of difficulty in dealing with such an apparent simple issue may have increased due to the differences in plants, diversity of utility management and the changes in regulatory requirements over the years since the first plants were built in the late sixties. For example, it wasn't until early 1970 that the Quality Assurance Program requirements became codified. Therefore, there is little or no documentation of environmental qualification other than the purchase orders describing the equipment as the best available at the time of purchase. On the other hand, newer plants may need to go beyond reasonable assurance of qualification with expensive programs detailing the paper trails left during the manufacturing and testing of purchased electrical equipment.

Making judgment on whether safety was substantially increased due to almost six years of hassling over the EQ programs is even more difficult. Are, for example, the environments created by the design basis accidents really the ones that endanger equipment? The TMI accident showed that an explosion of hydrogen due to an accident beyond the design basis can severely damage equipment within the containment. On the other hand, the equipment damaged was not required for mitigating the accident (e.g. the overhead crane). In the absence of really knowing the details of the manner in which accidents progress or there likelihood, perhaps the only approach is the one examined here.

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SHIFT TECHNICAL ADVISORS AND SENIOR OPERATORS

1.0 INTRODUCTION

1.1 Purpose

The purpose of this paper is to describe the Nuclear Regulatory Commission's requirements on assuring there are sufficient numbers of qualified operators to operate a nuclear power plant. These requirements which were changed following the accident at TMI-2 are but one small part of an overall NRC program to consider in depth the effect of humans on the safe operations of nuclear power plants. This program is referred to as the NRC's Human Factor Program and encompasses such topics as control room designs, plant operating procedures, operator licensing and man-machine interfaces. This larger program is still developing. In this paper, only the area of operator staffing will be examined.

1.2 Background

A nuclear power plant's operating organization consists of several departments with management supervisors and workers. Generally, the departments are maintenance, operations, radiation health and chemistry and technical support. The plant must be manned 24 hours a day and many of these departments must provide workers on shift depending on the workload, power requirements, and the plant's "mode" (e.g. 100% power, refueling, shutdown). The department in the plant's organization that requires all of its workers and many of its supervisors to be on shift is operations. Regardless of the plant mode, operators and their supervisors must be there to monitor the plant, perform testing, maneuver the reactor, etc.

The operating department generally consists of an operating supervisor that reports directly to the plant manager and "shifts" of operators that report to the operating supervisor. Each shift consists of a shift supervisor, reactor operators, and roving operators (floor men). The number of shifts needed for 24-hour coverage providing each shift works 8 hours a day is generally 5. Three for each 8-hour shift in the day, one for "days off" and one for "sick leave". The added burdens of training and extra work during startups and shutdowns require either additional shifts or the operators to work longer hours.

1.3 Operator Licensing

The NRC's rules which govern the staffing of nuclear power plants with NRC licensed operators are contained in three paragraphs of the code under a lengthy section dealing with the issuance of operating licenses and construction permits to utilities. (10CFR50.54, (j)(l)(m)). Under the sections, a licensed operator or senior operator must be at the controls at all times during operation of the facility. A licensed senior operator must be readily available on call or be present at the facility.

The licensing of operators under the NRC's rules is contained in a separate section of the code titled Operators' Licenses (10CFR50 part 55). It is in this section that the distinction is made between a "senior operator" and an "operator". A licensed senior operator may direct the activities of licensed operator and manipulate the controls of the reactor. A licensed operator may only manipulate the controls.

In order to become a licensed operator, a person must be enrolled in a formal training program usually run by the utility (sometimes with contracted help) and take an NRC administered examination. The examination consists of a written exam that takes about one day and an oral exam conducted by an NRC examiner during a "walkthru" of the plant. Successful completion of both the written and oral exam is required in order to be licensed by the NRC.

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As mentioned above, a licensed senior operator may supervise the activities of licensed operator. Therefore, the exam for a senior licensed operator will include topics that deal with greater understanding of the design basis of the plant and administrative procedures than found in the operator's exam. Also, a prerequisite for sitting for the senior exam is the successful completion of the operator's exam.

Therefore, when a utility begins to operate a plant, it must have in place sufficient number of licensed people to be assured that there is at least one licensed operator in the control room and at least one senior operator to supervise. Many plant operating licenses contains administrative requirements which include the rest of the shift compliment of non-licensed roving operators and tables describing shift staffing requirements for special plant conditions such as refueling.

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2.0 HISTORY

2.1 TMI Events Related to Shift Staffing

The above requirements on staffing and qualifications were in place prior to the accident at the Three Mile Island Unit 2 plant. Following the accident, several investigations revealed weaknesses in the operator qualification and shift staffing. The weakness in qualifications appears to have focussed on the absence of a college degreed engineer on shift. The weakness in staffing centered around an apparent loophole in the rules that would permit the licensed operator to be unsupervised in the control room by a licensed senior operator.

The weaknesses can be traced to two events that occurred during the TMI-2 accident. The first is that the preponderance of control room indications during the first two hours of the accident from exit core thermocouple readings to readings of temperature and pressure in the reactor coolant system should have told the operators that there was insufficient heat removal from the core and that core damage was imminent. As a result of that observation and the absence of a degreed individual on shift, a degreed shift technical advisor (STA) was added to the existing staffing requirements.

Also, during the accident, the licensed senior operator in charge of the shift (usually referred to as the shift supervisor) left the control room to supervise and assist the floor operators with a problem in the secondary plant that started the accident sequence. Many of the investigators felt his absence from the control during such a critical time may have contributed to the subsequent events that lead to a damaged core. As a result, consideration was given to the addition of another senior operator to the shift with the sole responsibility of directing the activities of licensed operators from the control room.

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2.2 The Shift Technical Advisor

A few months after the accident (July 1979), an NRC task force issued a report recommending a variety of lessons learned from their investigations of the accident and the responses obtained from licensee from the initial series of bulletins and orders sent out immediately after the accident. The report is titled the TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations (NUREG-0578). It is in this report that a requirement for a shift technical advisor is first mentioned.

The role of the shift technical advisor (STA) was to provide the licensed operators with "engineering expertise" obtained from the STA's participation in a formal degree program in science or engineering. This role was advisory only as the STA reported to a department which was not responsible for day-to-day operations. In the event of a transient or accident, however, the STA "would report to the shift supervisor in the control room".

Prior to being placed on shift, a utility run training program was developed which schooled the STAs in plant specific transients and accident analysis. The qualification of the STA was the sole responsibility of the utility with NRC review of the utility program.

The implementation date for this requirement was January 1, 1980. Operating plants were sent orders and plants nearing issuance of an operating license were told by the NRC to add the requirement to their draft operating licenses.

2.3. Shift Manning

The first mentioned of an increase in the number of licensed operators in the documents released by the NRC related to the TMI accident is in a

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report titled "NRC Action Plant Developed as a Result of the TMI-2 Accident". (NUREG-0660). The report was published in May 1980. The plan was developed to "provide a comprehensive and integrated plan for actions now judged necessary by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident." The Action Plan, as it became referred to, was an attempt by the NRC staff to categorize, cross-reference and schedule all the recommendations from the Kemeny Report, The Rogovin Investigation and early NRC staff studies of the accident. The shift manning requirement came under task I.A. "Operating Personnel" and was given the number I.A.1.3. This requirement stated that number of operators to be present in a control room shall include a licensed reactor operator and a senior licensed reactor operator at all times during reactor operation. The implementation of this requirement was scheduled for July 1, 1982.

2.4 Clarification of the Requirements

On October 31, 1980, the NRC staff issued letters to utilities which clarified the requirements in NUREG-0660, the TMI Action Plan. This clarification results from a series of meetings held with the industry in the NRC's regional offices in which the NRC staff provided more detailed explanations of the Action Plan requirements and the industry provided comment. The results of these meetings were published as NUREG-0737 "Clarifications of TMI Action Plan Requirements" published in November of 1980.

The STA function (Action Plan item I.A.1.1) was not changed from the previous guidance in NUREG-0578. The clarification of the STA requirement appears to focus on the criteria a utility must use to select, train and qualify on STA. The staff referred to a INPO document entitled "Nuclear Power Plant Shift Technical Advisor, Recommendations for Position Description, Qualification, Education and Training." The

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staff further notes that sections 5 and 6 of the INPO document are an acceptable approach for the selection and training of STAs but that the INPO document should not be considered as a requirement but only as "interim guidance". The schedule for completions of the STA program remained unchanged from NUREG-0578.

The new shift manning requirement as stated in Action Item I.A.1.3 were more clearly stated in a July 31, 1980 to utilities "stating that a licensed senior reactor operator shall, at all times, be in the control room from which the reactor is being operator". A chart delineating the shift manning requirements for dual and multiple unit control room was attached.

In NUREG-0737, the requirement for a senior operator to be present in the control room was not changed from the requirements as stated in the Action Plan or a July 31, 1980 letter clarifying the requirement. Much of the clarification in NUREG-0737 dealt with the use of overtime and finalizing requirements on overtime in an NRC bulletin 82-02 "Nuclear Power Plant Working Hours".

2.5 Drafting of the Shift Manning Rule

On June 11, 1982, the Commission held a meeting to discuss with staff the ability of utilities to meet the shift staffing requirement of NUREG-0737. As a result of that meeting, the Commissioners directed the staff to write a rule implementing the previous staff manning requirements of NUREG-0737. The Commissioners approved publication of the proposed rule with a 30-day comment period on July 29, 1982 and the proposed rule was published on August 30, 1982.

On February 7, 1983, the Commissioners were sent a paper (SECY 83-52) seeking their approval of the final rule. The period between publication of the proposed rule and the presentation to the full Commission in its

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final form included the time needed for collection and analysis of public comments as well as review by the NRC's Executive Director for Operation (EDO) Committee to Review Generic Requirements (CRGR).

CRGR met on January 6, 1983 to consider the final rule. The Chairman of the Committee disagreed with the Committee's decision to approve the final rule. His written dissent in the minutes of meeting states that the issue of plant staffing is a complex one, the cost outweigh the benefits, the issue of plant staff qualifications is still under review and the rule does not address the requirement for an STA. The STA requirement is handled thru a non-binding Commission Policy Statement attached to the rule. This policy statement permitted a utility to combine the second senior operator on shift with the functions of the STA provided the senior operator meet certain educational requirements. The CRGR Chairman does not provide a specific cost-benefit analysis of the rule to substantiate his statement in the minutes. These responses seem to parallel much of the industry comment on the same rule.

On February 17, 1983, the Commission met to consider the final rule and the policy statement on combining the senior operator for the control room and the STA as stated in SECY 83-52. That meeting resulted in several questions concerning the justification of the final rule, definition of the boundaries of the control room and accounting for sudden illness or other expected situations causing the absence of an operator in the control room. As a result, on March 14, 1983, the staff revised its previous paper to the Commission and sent a new paper (SECY-83-52A) to the Commission for consideration. On April 21, 1983, the Commission met again to discuss and approved the final rule. They did not, however, approve for publication in the Federal Register the policy statement on combining the second senior operator and the STA. The rule change was finally published on July 11, 1983 with an effective date of January 1, 1984. The policy statement has yet to be published.

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3.0 ANALYSIS

3.1 The Need

Nuclear power plants are complex machines that have a range equipment from extremely simple (such as float valves found in household toilets) to very complicated (such as the main generator for power production). The range of experience, training, and education required by the people that operate and maintain the equipment is equally as wide. A floor operator who takes direction from a licensed operator may not need to know as much as a licensed operator because the floor operator is generally under the direct supervision of the shift foreman who normally has a license. There is little or no disagreement that positions on shift at or below the foreman level do not require a degree.

The shift supervisor, however, is in a unique position between the plant's organizational positions that require a degree such as the plant manager or operations supervisor and the non-degreed union positions such as the shift foreman and the control room operator. If the shift supervisor is to be the plant manager's representative in the plant manager's absence, then the same qualification requirements for the plant manager should apply to the shift supervisor. On the other hand, the plant manager is not required by the NRC rules to have a senior operator's license. To complicate matters further, the qualification requirements for positions in a nuclear power that do require degrees are guides, not requirements.

In order to solve this dilemma, both the industry and staff have undertaken studies called job/task analysis of the shift supervisor's position to determine if a degree is required. The purpose here is to define each individual components of the shift supervisor's job, decide what training is needed for each task defined and then determine if the shift supervisor needs a degree. These studies may be completed by the end of 1984.

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A more easily understood need is the need to have adequate number of qualified operators on shift. An analogy can be drawn between equipment used to mitigate accidents and the people that operate the equipment. A nuclear power plant, for example, cannot operate at full power if the emergency diesel generators are broken. If the diesels fail while the plant is operating, then the reactor must be shut down (usually within 6 hours) and cannot start-up again until the diesels are fixed. On the other hand, there exists no similar requirement on operators. It is true that in order to run the plant at 100% power, there must be a senior operator in the control room as well as other licensed and unlicensed operators. However, the number of licensed operators is a requirement for <u>each shift</u>. Therefore, it is convinceable the requirement could be met with two shifts, working 12 hours a day. This would, of course, violate the NRC policy guidance on the use of overtime but that policy is not part of the code of federal regulations and is non-binding on the utility. It, therefore, appears that only a portion of the shift manning problem identified in the investigations of TMI has been solved. Until the NRC issues regulations on the total number of qualified operators such as those that apply to safety-related equipment, then the number of operators on shift may be adequate, but their alertness may be jeopardized by working successive 12 hours a day.

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4.0 CONCLUSION

If an operator were simply faced with the problem of mitigating design basis events, his job would be reasonably simple. For example, if the largest pipe in the reactor primary system breaks, the accident progresses quickly, safety system come on automatically. The operator's role is to monitor the equipments' performance. If a pump that was supposed to come on doesn't, then he would turn the pump on manually. He then becomes a backup to the safety actuation systems which were supposed to start the pump.

However, the large pipe break accident has never occurred. The events the operators usually face are multiple failures involving both safety and non-safety related systems. The TMI accident started out as a loss of feedwater accident (a design basis transient) but quickly turn into one involving multiple mechanical failures coupled with human errors. In such circumstances, it is very difficult to lay down procedures telling the operator specifically what to do for each of these conceivable events. It, therefore, seems important to give the normally non-degreed shift supervisor the analytical ability such as shift technical advisor to analyze those events that are beyond the design basis or the operating procedures. On the other hand, it is unreasonable to expect anyone, whether he has a degree or not, to analyze in a few minutes several hundred alarms, printouts, and meter readings and to decide what course of action should be taken in order to terminate an event safely.

The problem is further complicated by the unusual relationship between the shift supervisor and the STA. Under normal operating condition, STA works for the on-site safety group but during upsets, he "reports" to the shift supervisor. During upsets, he is only an advisor and not directly responsible for the outcome of the accident. Therefore, there is really no incentive for the shift supervisor to listen to the STA who is not responsible by virtue of not being a licensed senior operator. It is perhaps this dilemma that lead the NRC to permit utilities to combine the functions of the STA with the control room senior operator. This has been done but the policy statement has yet to be published and is not a requirement anyway but merely an option.

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Finally, it is not obvious to what extent the safety of the plant should depend upon the actions of the operator. Perhaps, the operator's role during a transient or accident is to let the safety systems function and keep his hands off the controls for a specific period of time. This would alleviate the need for instantaneous analysis of upsets and put more of the burden of making the plant safe on the plant's designer. This approach would, however, do little to help existing plants as extensive backfits might be required and there would always exist that one accident the designer had not considered. As difficult as this problem is, the regulatory system at least solved portions of the problems discovered from investigating the TMI accident and has placed more emphasis on the role of people in nuclear power plants, both in the industry thru INPO and in the NRC thru the Human Factors Program.

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NUCLEAR REACTOR REGULATION

AND

NUCLEAR REACTOR SAFETY

Submitted to:

Energy Program
Office of Technology Assessment
Congress of the United States
Washington, D.C. 20003

Under Contract No.: 333-7880.0

Submitted by:

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September 6, 1983

SC&A

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1.0 INTRODUCTION AND SUMMARY

1.1 Background

This study was conducted for the Energy Program of the Congressional Office of Technology Assessment (OTA) as part of an OTA assessment entitled "Nuclear Power in an Age of Uncertainty." The purpose of this study was to analyze the relationship between the nuclear reactor regulatory process and nuclear reactor safety. It will be used to complete a chapter of the OTA report which deals with the regulation of nuclear power. Most of the work was performed, and this report was written during the month of August 1983.

1.2 Approach

The study consists of two parts -- a survey of informed attitudes toward the regulatory process and an analysis of two case examples.

The survey was performed by means of telephone interviews. The number of people interviewed was small -- three nuclear critics, one reactor vendor senior manager, one architect-engineer senior manager, two electric utility senior managers, two members of the Advisory Committee on Reactor Safeguards, one Nuclear Regulatory Commission licensing project manager, and one Nuclear Regulatory Commission senior manager. While this represents a very small sample of those concerned with nuclear power, the individuals interviewed were chosen based on their qualifications to address the issues. The method used in the survey was to ask each individual a number of questions (in most cases, copies of the questions were provided to the respondents prior to the actual interviews). A list of the questions is provided as an appendix to this report. Each interview took approximately one and one-half hours. A summary of the survey results is given in Chapter 2.

Two case examples of recent regulatory requirements were examined in order to illustrate through example how the system actually works. The two examples selected -- fire protection and ATWS (Anticipated Transients Without Scram) were chosen both out of expediency (ready availability of documentation) and in order to illustrate two significantly different issues. Condensed

chronologies of these case examples are annotated and summarized in Chapter 3. Much of the descriptive material contained in Chapter 3 was excerpted or taken directly from various Nuclear Regulatory Commission regulations, dockets, and reports.

1.3 Summary

From the nuclear critics' standpoint, the rules are in place, but are not specific enough and are not being properly enforced. From the industry's standpoint, the rules are overly specific, and result in an enormous paperwork burden that diverts resources, some of which could otherwise be used to improve safety. The Nuclear Regulatory Commission (NRC) staff is concerned about the consistency of its reviews, pointing to a lack of management direction and differences among regions. Staff members also cite the diversion of staff resources toward work that is not truly safety-related as being a problem. Members of the Advisory Committee on Reactor Safeguards believe that priorities are not being determined properly, if at all, and that the interactions of system changes with other parts of the nuclear power plant are not being adequately addressed.

It seems probable that at least some of the perceived difficulties with the regulatory process stem from the development of that process at the same time that the industry was undergoing rapid growth. "Expertise," both in building and operating increasingly larger and more complex plants and in learning to regulate them safely and effectively, had to be developed essentially by "on the job training." Certainly in hindsight, this type of parallel development does not foster the most self-consistent regulatory system. In fact, a close parallel may be drawn with environmental regulation over the past 15 years. The case examples support this viewpoint, illustrating substantial reliance by the AEC/NRC on the industry for analyses, inordinately long periods of time to resolve issues, and the lack of anticipation of potential problem areas.

A number of changes could be made to improve the efficiency of the regulatory process, though it is not apparent that any of these would result in significantly safer reactor operation. NRC should be encouraged to develop

greater in-house expertise so that reliance on industry for analysis can be reduced. An attempt should also be made to increase the clarity of rules, regulations, and technical positions. Even though it might reduce the number of public access points into the licensing process, the proposed one-step licensing process has several advantages as long as agreement can be reached on the definition of a final plant design. Certainly a one-step process would reduce the amount of inefficiency and redundancy in the existing two-step process.

Backfitting appears to have become more of a political than a technical issue. The case examples illustrate that considerable technical knowledge has been gained over time, even on issues that are still officially unresolved. It may be that the technical knowledge acquired and the analytical tools developed are finally sufficient to deal with any new problems discovered, thus permitting the adoption of a sensible and widely accepted backfit procedure.

The area of most extensive agreement among all parties is that enforcement should be stronger and more uniform. There are considerable advantages to an improved enforcement policy for all parties. Obviously, the industry is not willing to risk another accident like Three Mile Island. The existing enforcement policy does not weed out the consistently poor performer.

Although the existing regulatory system has a number of deficiencies, all parties have learned to work within the system and there are no obvious changes that would substantially increase the level of safety. The existing system has been defined by roughly thirty years of turbulent evolution. Indeed, the aftermath of the accident at Three Mile Island may have removed some of the final faults from the system. Fine tuning, with particular attention to issues of public safety and perception, appears to be a more productive path than a major overhaul of the existing regulatory system.

2.0 SUMMARY OF THE SURVEY RESULTS

This section summarizes the results of interviews with three "nuclear critics," four members of the nuclear "industry" (one reactor vendor senior manager, one architect-engineer senior manager, and two electric utility senior managers), two members of the "Advisory Committee on Reactor Safeguards," and two staff members of the "Nuclear Regulatory Commission" (one licensing project manager and one senior manager). In the summary that follows, we refer to the respondents generically, by the names given above in quotes. Although the interviewees were chosen based on their qualifications to address the issues, our results are based on a very small sample indeed, and the generic designations are used only for ease of presentation.

2.1 Faults in the Regulatory Process

Institutional Concerns

Under the current system, nuclear critics can raise safety concerns at any time: during hearings, by telephone or letter to either the Commission or the staff, and by petition. Theoretically, therefore, there are a number of avenues available. From the critics' point of view, however, they are dealing with a system whose collective mind is already made up by the time they get to participate. The nuclear critics cite several issues they have raised which were ignored by the Nuclear Regulatory Commission until either through sheer persistence on the part of the critics, or by the occurrence of an event, the critics' views were finally given attention.

The nuclear critics view the Nuclear Regulatory Commission as an agency which, after having reviewed the application for, say, an operating license and having reached agreement with the applicant over issues in contention, will vigorously defend its position, brushing aside the critics' complaints. They question the impartiality of the hearing boards, stating that few, if any, of the board members approach the hearings prepared to reject a license application. Another related point raised by the nuclear critics is that the design process for a nuclear plant is not complete at the construction permit

stage, so that their ability to make constructive criticisms at this stage is limited.

On the other hand, at the operating license stage, the final design is complete, it has been reviewed by the Nuclear Regulatory Commission staff, and the plant is virtually built, so that the critics' concerns are directed toward a group that has a vested interest in defending its position that the plant is satisfactory. They contend that the path by which the Nuclear Regulatory Commission staff reaches its conclusions with regard to safety concerns is not traceable. That is, the staff's reasoning process is not written down so that interested parties are able to understand what the problem is and how the staff arrived at its recommended solution. There is also concern about the independence of the licensing boards from the Commissioners (who can overrule decisions) and the recent trend toward the use of contractors to do reviews formerly done by the staff, without adequate quality control by the staff. (Note - the Nuclear Regulatory Commission does not have any separate internal quality assurance or quality control organization.)

The nuclear critics view the industry as being more concerned with profits than safety, thereby meeting only the minimum safety requirements set by the Nuclear Regulatory Commission. They believe that the Nuclear Regulatory Commission caters to industry, allowing them to negotiate lesser requirements than are necessary and to implement these requirements on a protracted schedule. The critics consider the Commission's decision to remove financial qualifications from licensing as being incorrect in view of the recent financial troubles of some utilities. They believe that a financially troubled utility will not be able to devote proper resources to safety.

According to the industry, the principal problem with the current process is lack of management control at the Nuclear Regulatory Commission. This is reflected in uneven reviews, a lack of priorities, and extensive and not-well-thought-out modifications which have made plants more complicated, less predictable, and therefore quite possibly, less safe. Some utilities feel that they get rewarded for delay and punished for taking early action, noting that they have been forced to start over and re-do what they considered

to be adequate solutions to a safety concern, while others, who delayed until the Nuclear Regulatory Commission made up its mind, were granted relief from the schedule and sometimes from the requirement.

The industry maintains that small problems are given the same degree of attention as large ones, and in a related vein, that reporting requirements call for so much detail that resources are diverted from significant issues to insignificant ones. The Nuclear Regulatory Commission on the one hand studies some small problems to death, while on the other makes snap judgments on large issues which later have to be revoked. The general industry comment in this area is that there does not appear to be any true decision-making process, and that it should be the function of the Nuclear Regulatory Commission management above the branch chief level to set priorities in a rational way, not just to listen to the loudest voice.

According to the industry, good utilities, who maintain complete records of their problems and solutions, thereby creating a traceable audit path, get more bad marks from the enforcement staff than the utilities who maintain poor records, thereby not providing the staff much to criticize.

The industry maintains that hearings have not identified any significant safety concerns, contending that it is the interaction between the industry and the staff that accomplishes this function and that both industry and Nuclear Regulatory Commission staff resources are wasted. The nuclear critics disagree and note that the very existence of the hearings forces both the staff and the industry to do a better job than they would otherwise do.

According to the industry, the nuclear critics are unwilling to admit that every human endeavor involves some risk and that the risk from nuclear plants is lower than or comparable to the risks from alternative means of generating power. The industry feels that few of the critics are knowledgeable, that they tend to deal in slogans rather than facts, and that they continue to raise issues which have been resolved to the satisfaction of the Nuclear Regulatory Commission as a means of delay, rather than from any real concern.

The industry responds to the critics' concern that raising of issues at the operating license stage is too late by pointing out that the critics can identify their concerns to the Commission at any time. The nuclear critics, on the other hand, believe that the industry puts profits before safety when they set priorities, and that competition produces a minimally safe design.

The Nuclear Regulatory Commission staff views itself as being too closely watched over and having too much of an adversarial relationship with the Congress. Staff members feel they are burdened with an inordinate amount of paperwork, which does not allow them time to concentrate on their mission of safety. They view the regulatory process as too cumbersome and too legalistic in an area that is primarily technical. They do not view themselves as being in league with industry; however, they do believe the Commissioners are often not forceful enough in dealing with the poorer performing utilities.

In the area of licensing, Nuclear Regulatory Commission personnel see a problem with the consistency of reviews. They feel that the standard review plan helps to even out their reviews, but believe there is still a problem in balancing the benefits of dissent against the need for reasonable management control of the review. In a related vein, they view the trend toward putting more staff functions into the regions as having a negative impact on consistency.

The staff does not believe hearings contribute very much to safety. Staff members point out that by the time hearings are held for an operating license, they are convinced the plant is safe because all but a few of the issues have been resolved. Staff members reply to the criticism that they enter the hearings with a closed mind by pointing out that they have reviewed the plant in great detail and know much more about it than the critics. On the other hand, they point out that their review is, of necessity, an audit review; they further note that the ratio of the hours spent in design to those spent in review is on the order of 10,000 to one.

The staff views the so-called Sholly Rule, which requires that a notice be put into the Federal Register before making any change to a plant's technical

specifications, no matter how trivial, as a huge drain on staff resources with no accompanying benefit to the public.

The Nuclear Regulatory Commission staff believes that the industry attitude toward safety has improved over the past few years. Staff members believe that top utility management for the most part is now committed to safety, and that ultimately the responsibility for safety lies with the utilities.

Members of the Advisory Committee on Reactor Safeguards are concerned that the Commission's rules are poorly worded and do not always say what they mean. Problems arise when inspectors in the field have to enforce a rule based on what was written, not what was meant. Individual judgment on the inspectors' part as to whether the intent of the rule is being met is discouraged on the legalistic basis that should the matter end up in court, it cannot be shown that the utility may have acted in bad faith.

Committee members believe that the utilities have the ultimate responsibility for safety, and that the system should recognize this by rewarding good performance, and penalizing bad performance. They consider that even the most competent and effective Nuclear Regulatory Commission could not make an incompetent or unwilling utility operate safely if the utility did not accept this responsibility (short of shutting down the plant). The members are concerned about the lack of standards for judging whether a utility starting its first plant is competent and the lack of performance standards for operating utilities.

Members of the Advisory Committee on Reactor Safeguards do not believe that hearings are the place to raise safety concerns; the hearing process rarely resolves anything. They consider that the principal forces driving safe plant operation are events and operating experience.

Backfitting*

The industry does not consider that the large number of changes made to plants over the years has contributed much, if anything, to safety. Their belief is that safety must be designed and built into the plant from the beginning, and that plants have become more complex not because they are larger, but because of the many changes dictated by the Nuclear Regulatory Commission. They consider that these many changes have made the plants more complicated, less predictable, and possibly less safe.

The industry has the view that the Nuclear Regulatory Commission's requirements are overly prescriptive. They note that frequently the Commission has had to change many requirements because they simply did not work, and that this is cited by the nuclear critics as evidence that the Commission and the industry are in league with one another. The industry contends that the rules should be written as criteria, with each plant determining how it can best meet the criteria, subject to concurrence by the Nuclear Regulatory Commission.

The industry feels that most true safety concerns are generic; that there are few concerns that are so specific that they would not be concerns at other plants as well. Thus, the industry contends that general criteria are preferable to prescriptive rules. Moreover, by treating problems generically, the industry believes that owners' groups can be set up to solve them more effectively.

A criticism of this approach is that with 70 plants, there could be as many as 70 different solutions to the safety concern, thus creating a severe resources problem for the Nuclear Regulatory Commission in its review. The industry's response to this is that the staff should audit the solutions against the guidelines, giving the most attention to the least capable utilities.

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*Backfitting as used here includes "ratcheting," which usually refers to changes required after a plant goes on line.

Another problem cited by the industry with respect to changes imposed by the Nuclear Regulatory Commission is its failure to set priorities; that lesser concerns receive as much attention as greater concerns. Industry views the recently established Committee to Review Generic Requirements (CRGR) as a positive step, but is somewhat skeptical as to how well it will work. A related issue is the use of engineering judgment versus probabilistic risk assessment (PRA) in setting priorities, and determining whether proposed changes are truly effective and worth the cost. The industry, nuclear critics, and Nuclear Regulatory Commission personnel all agree that both methods have their place, and that neither should be used exclusively.

The industry considers that the loudest voice at the Nuclear Regulatory Commission drives the system; consequently priorities for backfits are not being properly set. They concede that in the past the industry has been recalcitrant on some issues, and that this has led to the Nuclear Regulatory Commission's being overly prescriptive and establishing what appear to be unrealistic priorities and schedules. The industry considers that this situation has improved.

The nuclear critics argue that the current backfit rule is satisfactory;* the problem lies in their implementation and in the fact that the processes used to determine when and how to backfit are obscure. They would like to see these processes written down so that interested parties can examine, understand, and criticize them.

The nuclear critics contend that new safety concerns are simply added to a list instead of getting the attention they deserve. Nuclear Regulatory Commission personnel disagree with this, stating that the lists are reviewed periodically to make sure that resolution is proceeding. Members of the Advisory Committee on Reactor Safeguards are concerned that the Nuclear Regulatory Commission is weak on follow-up of these items and has no central organization to assure that none is overlooked.

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*It may be of interest to note in passing that the explicit backfit provisions in 10 CFR 50.109 have never actually been invoked.

The critics believe that more is claimed for probabilistic risk assessment than is deserved, but do agree that it can be useful is helping to determine priorities [Note - experts in the field of probabilistic risk assessment generally agree that relative, or comparative risk assessments are more accurate than absolute risk assessments; that is, an evaluation of the relative change in risk is more accurate than that of the absolute magnitude of the risk.]

The nuclear critics also believe that it is better to reassess the fundamentals of the design than it is to fix safety concerns with backfits. [n.b., the industry agrees with this comment, as noted earlier.] The critics find it ironic that the industry, while arguing for more formal procedures on backfits (documentation of both need and establishment of priorities), is asking for less formality in the licensing process.

The Nuclear Regulatory Commission staff considers that on balance, backfits have improved safety and that the process for determining priorities has improved greatly since the establishment of the Committee to Review Generic Requirements. Still, the staff believes that there is a tendency to expand requirements and strong management control is needed to assure that the costs imposed are justified by the benefits. In this regard, the industry considers that the Nuclear Regulatory Commission has done a poor job in estimating cost.*

Members of the Advisory Committee on Reactor Safeguards consider that the Nuclear Regulatory Commission has not done very well in assessing the interactions of backfits with the rest of the plant and believes that in more than a few cases the change as originally proposed would have caused (and in some cases did cause) a reduction, rather than an increase in safety.

Enforcement

Neither members of the Advisory Committee on Reactor Safeguards, the nuclear critics, the industry, nor the Nuclear Regulatory Commission staff believe

*This criticism has been addressed recently by the establishment of an independent office to address the costs of regulatory requirements.

that the current system of imposing fines is an effective means of enforcement. The nuclear critics consider that fines are simply added to the utility's rate base and hence have no significant effect, although the publicity is good.

The others point out that fining and chastising a utility in public before giving it a chance to work out a solution to the problem merely forces the utility to vigorously defend its position, a result that is counterproductive to resolution of the problem. Members of the Advisory Committee note that the press release describing the fine always indicates there was no risk to the public, a blatant inconsistency.

Nuclear Regulatory Commission personnel believe that enforcement is not as uniform as it should be, varying somewhat from plant to plant and from region to region. According to industry, utilities which have a good system for detecting safety concerns are penalized for finding and reporting them. Utilities which do not have a good system and therefore do not find safety concerns are not penalized. The industry considers that the system should differentiate between safety concerns resulting from random human failings and those resulting from a lack of attention on the part of a utility. The former should not receive much of a penalty (unless it is indicative of the latter), while the latter should receive vigorous enforcement action. Another industry complaint is that the Nuclear Regulatory Commission's reporting requirements are so extensive that they divert resources and can cause significant concerns to get lost in the noise. Nuclear Regulatory Commission personnel agree with this.

The nuclear critics believe that Nuclear Regulatory Commission inspectors are frequently overruled by their management who water down their reports. Another critics' complaint is that the Commissioners do not always support enforcement of the rules. The critics' view is that the industry will only comply if they are certain that enforcement action will be taken unless they do. The industry agrees with this view, saying that enforcement must be uniform and inevitable, and that the cost of non-compliance should be greater than that of compliance.

2.2 Suggested Improvements in the Regulatory Process

There appears to be a consensus that, while certain aspects of the regulatory process could be improved, radical changes to the system are not needed to improve safety.

In two areas there was agreement among all of the groups on how to improve the current process. These were one-step licensing, and sure and certain enforcement action for utilities failing to meet the minimum standards set by the Nuclear Regulatory Commission.

With one-step licensing both the design and the review by the Nuclear Regulatory Commission would be substantially complete before any hearings and before the start of construction. Modifications to the design would not be allowed unless a generic safety concern of substantial magnitude were to arise. Operation of the plant would then be contingent only upon the Nuclear Regulatory Commission's determination that the plant was properly built. The critics and the industry differ as to what constitutes a substantially complete design, the number and type of public hearings, and whether all issues have to be resolved before construction is begun.

Sure and certain enforcement of Nuclear Regulatory Commission rules is viewed by all parties as being necessary. The industry states that they cannot afford to have "weak sister" utilities, and that the only way to assure this is to have the Commission enforce its rules, including shutting down plants until safety concerns are resolved.

The industry would like to see the Institute of Nuclear Power Operations (INPO) used as a means to encourage the utilities to pursue a standard of excellence that is almost out of reach and significantly higher than the Nuclear Regulatory Commission's standards. The utility should never be able to satisfy these Institute standards, but should always be required to satisfy the Commission's standards.

The industry and the Nuclear Regulatory Commission staff both consider that before fines or other punitive measures are imposed on a utility, senior

representatives of Nuclear Regulatory Commission management should meet with the chief executive officer, inform him of its concerns, and attempt to reach a resolution on how to solve the problem and the schedule for the resolution. Only if that step were ineffective should enforcement action be taken, such action to include shutting down the plant if that were deemed appropriate. This approach would, in the industry's and Commission personnel's view, alleviate the current problem in which the utility must react defensively because of public disclosure before they have had a chance to propose action on the problem. Both industry and the Nuclear Regulatory Commission personnel agree that for this to work, enforcement would have to be viewed as certain if the utility did not act. In a related vein, the industry believes that the Nuclear Regulatory Commission should deal strongly with a utility which cannot or will not perform up to the minimum standards. The industry also considers it important for the Commission to be able to differentiate between "cannot" and "will not," because the enforcement action could well be different for each case.

The industry considers that whenever possible, safety concerns should be treated generically. Generic treatment would allow the industry to set up an owners' group and the Nuclear Regulatory Commission to assign the problem to a particular branch or task force. Both groups could then work together to resolve the problem generically.

The industry and Nuclear Regulatory Commission personnel agree that the "living schedule" approach currently being tried appears to be a good approach to the backfit problem. With this method of approaching backfits, the utility schedules the work over a five year period, assigning it on the basis of the priority, the outage schedule, and available resources. The resulting schedule is then reviewed and approved by the Nuclear Regulatory Commission.

The nuclear critics recommend that the Commissioners' power over the hearing boards be reduced; specifically that they not be allowed to tell the boards what issues they cannot consider. They would like to see the reasoning used in determining whether backfits are necessary, how they are to be implemented, and what the priorities should be, written down so that they can understand and criticize how the decisions were made. They would also like to have the

financial qualifications restored as one of the considerations in the licensing process. They also proposed that the hearing boards receive funding to hire their own independent experts to examine issues raised at hearings. This would make the boards less dependent upon the testimony of Nuclear Regulatory Commission staff and industry witnesses.

Other suggested improvements by the nuclear critics include more precise direction from the Congress to the Nuclear Regulatory Commission, and from the Commission to industry (more prescriptive, rather than less prescriptive rules), and some provision for funding of the nuclear critics to encourage more public participation.

Members of the Advisory Committee on Reactor Safeguards would like to see the system become less adversarial, so that more resources are devoted to safety and fewer to argument. They believe that sure enforcement of the regulations by the Nuclear Regulatory Commission is required. They also believe that the plant should be fully staffed well before it is ready to operate, so that operating and maintenance personnel are completely familiar with it.

There were two novel suggestions from industry for improving the system. In the first, the Price-Anderson Act would be repealed and the utilities would obtain all of their liability insurance commercially. The insurance companies, for their own self interest, would require the utilities to meet high standards. The Nuclear Regulatory Commission's role would be to assure that the insurance companies were doing a proper job of auditing the utilities for safety. The utilities, in order to keep their insurance rates at a reasonably low level, would voluntarily make improvements. Thus the system would be driven by economic, rather than political, forces.

The second suggestion would have proposed backfits reviewed by experienced engineers from industry, but not from the organization which did the design. Admiral Rickover used a similar approach in having each of his two laboratories (the Bettis and Knolls Atomic Power Laboratories) review one another's designs. This was an effective approach, as each laboratory knew the other was both competent and uninhibited in its criticism.

3.0 CASE EXAMPLES

A particular scheme for enhancing the safety of nuclear electricity generation through regulation may appear simple enough on paper. But in actual practice, whether to proceed through a formal rulemaking procedure, to issue show-cause orders to individual licenses, to request information for further staff consideration, or to develop staff positions that function as quasi-regulations are only some of the possible paths through the regulatory process. It has not been at all clear which path would have been proven to be most effective in any particular circumstance.

The two following case examples illustrate some of these points. Each example begins with a brief summary of the main issues raised in the regulatory process. The origin of the "problem" is then described and the actions taken to date are set out, including comments where appropriate. Much of the following was excerpted or taken directly from various Nuclear Regulatory Commission (NRC) regulations, dockets, and reports.

3.1. Fire Protection Regulations

Summary

Despite the fact that more than eight years have passed, this issue is still only partially resolved. Continuing and shifting differences of opinion within the NRC staff and between the ACRS and the staff, and a "legalistic" style of communication between the Commission, staff, and the Advisory Committee for Reactor Safeguards (ACRS) are apparent. Differences also exist in, for example, the question of whether formal rulemaking or show-cause orders to individual licensees was the better way to proceed. The staff also considered whether information requests and development of branch technical positions might form a regulatory position strong enough to support inspection and enforcement actions. An intervenor group provided fairly early pressure on the NRC, and the industry fought both the proposed and final rules in court.

Background

On March 22, 1975, the worst fire to date in an operating commercial nuclear power plant occurred at the Tennessee Valley Authority's Browns Ferry plant. The fire was eventually controlled, no radioactive material was released to the environment, and the reactor was safely shut down. However, many of the systems relied on for shutdown of the reactor under both normal and emergency conditions were not available because of extensive damage to control cabling of redundant systems. Although the fire was not particularly large or serious in terms of insurance underwriting loss, it was very serious in terms of the type of equipment that was rendered inoperative. Also of concern following this fire was the attitude and belief among those responsible for managing the fire that water should not be used on burning electrical cables because the resultant electrical faults would be worse than the fire damage. As a consequence, the fire burned for more than seven hours before it was extinguished with water.

Two recommendations made by the Special Review Group that investigated the Browns Ferry fire were intended to assure that the fire protection programs at operating nuclear power plants conform to General Design Criterion 3 (GDC 3), dealing with fire protection. One of the recommendations was that NRC should develop additional specific guidance for implementing GDC 3 requirements. The other was that NRC should make a detailed review of the fire protection program at each operating plant, comparing it to the guidance developed in response to the first recommendation.

Early Actions

In response to the first recommendation, the Office of Nuclear Reactor Regulation developed Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" (BTP 9.5-1), and Appendix A to BTP 9.5-1, "Guidelines for Fire Protection For Nuclear Power Plants Docketed Prior to July 1, 1976." BTP 9.5-1 was published for public comment in June 1976 as Regulatory Guide 1.120. Based on the comments received, the staff proposed extensive changes to the guide and presented the proposed changes to the ACRS in an open meeting in May 1977. Public comment was again requested, and

nineteen additional comment letters were also considered in Revision 1 of Regulatory Guide 1.120, which was published for a new one-year public comment period in November 1977. In general, the comments on Revision 1 of Regulatory Guide 1.120 were restatements of comments received during earlier comment periods and had already been considered and evaluated by the staff.

In response to the second recommendation (plant-by-plant review), the NRC requested each operating plant to submit a revised Fire Hazard Analysis. Most licensees proposed modifications to conform with the NRC guidelines. In conjunction with a site visit, the staff reviewed each licensee's analysis against the guidance contained in Appendix A to BTP 9.5-1 and made additional strong recommendations after each site visit. Licensees usually agreed to implement the recommendations.

Early in 1977, the staff considered various methods for enforcing compliance with the revised fire protection guidelines by November 1, 1980: (a) not permitting startup after that date if certain fire protection modifications were not complete, (b) setting a schedule for completion by license conditions, (c) ordering compliance under the provisions of 10 CFR 2.202 or 10 CFR 2.204, and (d) rulemaking. At that time, the majority of licensees were proposing modifications which would bring their facilities into compliance with the staff guidelines by the end of October 1980. Therefore, the staff decided to issue Safety Evaluation Reports to set the schedule for completing such modifications as conditions of the license. Most safety evaluations contained open items that were to be resolved on a schedule consistent with that set by the license conditions. In some cases, licensees agreed to further consider the need for recommended modifications and have since (1) adopted the recommendations, (2) proposed alternatives, or (3) continued to disagree with the staff recommendations and refused to adopt them. The staff decided most of these issues on a generic basis and licensees were notified. It appears that a number of licensees eventually made modifications in response to initial NRC recommendations, which later were held to be inadequate as the NRC staff position fluctuated.

The NRC review of these licensee submittals was at least a partial cause for some intervenor actions. On November 4, 1977, the Union of Concerned

Scientists (UCS) brought a petition before the Commission alleging that then current NRC regulations were not adequate to prevent destruction of redundant electrical cables. The petition was brought before the Commission rather than the staff because neither the Office of Nuclear Reactor Regulation nor the Office of Inspection and Enforcement nor had made the results of tests conducted by Sandia Laboratories in early July of 1977 available to currently sitting licensing boards by at least August of that year, nor had any corrective action yet been required of operating licensees. In essence, the Sandia tests (conducted in conjunction with NRC's Fire Protection Research Program) involved two redundant cable setups of seven trays each, conforming to current standards on flame retardancy, cable tray loading, and the physical separation distance of five feet. A fire begun in one tray of the simulated "safety" setup burned all the cables in those seven trays, propagated across the five-foot gap, and burned the cables in all seven trays of the "redundant" setup. UCS commented that this test showed that then-current licensing standards were inadequate because they could result in violations of general Design Criterion 3 of Appendix A to 10 CFR part 50 and other applicable NRC regulations and standards. Since BTP 9.5-1 was developed to provide more specific guidance for implementing GDC-3 in new plants and Appendix A to that BTP provided guidance for plants docketed before July 1, 1976, the UCS petition also cast doubt on future regulations insofar as they were based on BTP 9.5-1.

The Commission denied this UCS petition and a subsequent petition for rehearing filed in late 1977 or early 1978. The reasons for the denial were not available, but it appears possible from available documentation that the Commission felt that the revision of Regulatory Guide 1.120 issued in November of 1977 had sufficiently accounted for public comments, because responses to the revision seemed to differ little from comments on the original guide. However, the Sandia tests and their results are not mentioned in the SECY papers obtained for this review, and the Commission memo on the second UCS petition ordered the staff to proceed expeditiously in implementing the plan based on Appendix A to BPT 9.5-1.

However, the UCS petitions resulted in some activity within the NRC. On December 12, 1978 (more than eight months after the second UCS petition), the

NRR staff was requested to respond to questions raised by the UCS petitions about fire protection at operating plants (covered by Appendix A to BTP 9.5-1). The questions and responses are interesting because they point up the growing regulatory interest in the probability of events, demonstrate a careful and legalistic response pattern that still occurs, and do not touch the question of whether current standards were adequate. Three related questions and staff responses are summarized here:

Q: To what extent has the staff relied on probability in deciding that plants may continue to operate while the requirements of Regulatory Guide 1.120 are put into effect:

A: The staff relies only partly on the low probability of a serious fire occurring, relying also on the facts that the threat of a fire to public health and safety is indirect; at the beginning of this period [development of BTP 9.5-1 and Appendix A], existing plants had a fire protection program that met industry standards [no mention is made of the Sandia Tests] and programs have been improved since 1975. Continued operation is based on assessment of existing plant conditions, not on a calculated probability of fires.

Q: How can the staff rely on fire suppression systems in its probability judgment when the staff has required an alternate or dedicated shutdown mechanism after taking possible fire suppression into account in approximately one-half of the plants evaluated prior to July 1978.

A: The staff's judgment that the probability of a severe fire is sufficiently low to permit continued operation is based upon the opinion that effective administrative controls and fire mitigation measures will prevent a severe fire. On the other hand, in certain configurations the controls and fire mitigation measures may fail in some small fraction of instances. If, on the basis of conservative assumptions, the staff deems that a severe fire could damage redundant shutdown systems, an alternate or dedicated shutdown mechanism is required.

Q. Why is the staff reliance on probability not inconsistent [sic: should probably be "consistent"] with the Vermont Yankee case, which states:

"Turning to that situation, we rejected at the outset two of staff's arguments in support of continued facility operations. The first is . . . that there is a low probability of a loss of coolant accident in the time required for the reopened proceeding. That argument may be factually sound, but it constitutes an indirect challenge to the applicable criteria, in that it would permit licensing of a non-complying reactor. Consequently, we need not consider the factual question concerning the degree of probability of a LOCA in the next few months."

A. GDC-3, regarding fire protection, is quite different in form from GDC-35, relating to emergency core cooling following a LOCA. GDC-35 requires a specific system and, with the definition of Loss of Coolant Accident, specifies how large an event is to be considered (". . . up to and including . . . the double ended rupture of the largest pipe in the reactor coolant system . . ."). GDC-3, on the other hand, simply requires in rather general terms that "structures, systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room . . . ."

Consideration of the likelihood of fires of differing degrees of intensity at various locations in the facility having differing degrees of sensitivity with respect to overall reactor safety is an intrinsic aspect of applying GDC-3 to any particular nuclear facility.

The Proposed Rule

As stated above, the NRC staff had issued safety evaluation reports for each licensee following review of its fire protection programs. Most of these evaluations contained unresolved items. About 250 of those open items consisted of licensee commitments to provide further information or verification; another 250 consisted of missing information, further licensee consideration of NRC requirements, and late submittals. All through 1977 and 1978 the staff continued to develop acceptance criteria representing their recommendations for minimum requirements. When it became clear that some of these requirements would not be accepted by all licensees, the staff began action in December of 1978 to order compliance on the single specific issue of fire brigade shift size. Some licensees said they would have to be "ordered" to meet a five-man requirement. The NRC legal director took the position that rulemaking was more appropriate than compliance orders, as it exposed NRC to fewer adjudicatory hearings. It appeared a rule could not be available for publication until late 1979 or early 1980. Subsequently, the staff expanded the rulemaking to cover 17 generic unresolved issues.

On February 13, 1980, a paper was forwarded from the Office of Standards Development to the Commissioners to obtain a decision on whether rulemaking or orders should be used to enforce "minimum fire protection requirements" on plants in operation prior to January 1, 1979, so that all modifications would be in place by November 1, 1980. [Note that operating plants are the focus; plants in the licensing process appeared to be less of an issue. It is also noteworthy that the date of applicability to "operating" plants was changed from July 1, 1976 (Appendix A of BTP 9.5.1) to January 1, 1979, undoubtedly a major cause of concern to several licensees.]

Four major fire protection issues were stated in this proposed rule:

(1) The minimum fire protection requirement for redundant systems or components important to safety that are located in the same fire area.

(2) The different minimum requirements for systems and components associated with shutdown as opposed to loss-of-coolant accidents.

(3) The minimum shutdown capability required following a major fire that is not controlled by the automatic and manual fire protection features.

(4) All modifications related to fire protection shall be completed by November 1, 1980.

Appendix R to the proposed rule set out specific guidance on the 17 unresolved issues, condensed from the longer list originally included as Appendix A to BTP 9.5-1. At least three of these issues received more restrictive interpretation.

ACRS Response, the Revised Proposed Rule, and Licensee Actions

At some point between February and October 1980, the proposed rule appears to have been revised to permit exemptions from the Appendix (which listed the 17 specific criteria) for those fire protection "features" that were previously accepted as meeting the provisions of the Appendix to BTP 9.5-1, as reflected in the evaluation reports. A revised, "more relaxed, and substantially more realistic" schedule for implementation of modifications was also included, implying that the November 1, 1980 schedule was opposed by at least some licensees.

The ACRS considered the proposed (revised) rule at the early October 1980 meeting. The ACRS favored compliance orders over any rule, considering "the extremely prescriptive requirements of BTP 9.5-1" not the best means of ensuring safety, and holding that compliance orders would be more realistic in terms of backfit requirements, as engineering and design could be conducted under less time pressure.

While no specific dates are available, the nuclear industry brought a lawsuit to stay implementation of the almost certainly to be revised rule. This suit was probably filed in late 1980 or most likely in early 1981. The suit first

requested the Commission to stay implementation of the rule which was denied. Then the suit was brought before the circuit court, which upheld the NRC. Industry then appealed to the District court, which sustained the rule and the Supreme Court refused to consider industry's appeal. Therefore, as far as operating plants were concerned, the case was "settled."

Industry's arguments in attacking the proposed rule are rather interesting. First, industry asserted it could not comment intelligently on the rule's basis because the Commission had not identified that basis. Related to this was industry's contention that the technical basis did not in fact support the rule. Third, industry argued that the final version of the rule was so different from the one first proposed that the comment period should have been reopened. Finally, industry stated that the Commission had failed to follow its own backfitting regulation. (It also appears that NRC staff was split over whether backfitting should have been required.)

The rule (10 CFR 50.48 and Appendix R to 10 CPR 50), including exemption procedures, became effective on February 17, 1981. The rule required the licensees of all plants licensed prior to January 1, 1979, to submit by March 19, 1981: (1) plans and schedules for meeting the applicable requirements of Appendix R, (2) a design description of any modifications proposed to provide alternative safe shutdown capability pursuant to Paragraph III.G.3 of Appendix R, and (3) exemption requests for which the tolling provision of Section 50.48(c)(6) was to be invoked.

As discussed in the next section, a great many requests for exemptions were filed. It is likely that the liberal exemption procedure was a principal reason the rule was upheld in court. However, liberal use of exemptions can lead to less uniform enforcement than some other regulatory schemes, and exemptions may be more difficult to deny because licensees can request adjudicatory hearings, which are very costly in terms of staff time.

Implementation of the Rule for Operating Plants

By the end of September 1981, problems with implementing the rule were becoming apparent. First, the majority of the licensee responses to a February 20, 1981 request for information were not complete. When the inadequacies were explained to each licensee, commitment dates for revised responses were set. However, these dates did not allow the technical evaluation of the responses by the staff to be completed by Brookhaven National Laboratory on the original schedule. The staff therefore issued safety evaluation reports containing open items to be resolved after adequate responses from the licensees were received. This approach aided in identifying unresolved items, but it also demanded additional staff effort and put off resolution of Appendix R concerns.

The staff was also concerned that licensees would not make alternative shutdown modifications until NRC reviewed and approved the modifications. To counteract this, the staff intended to set a fixed date for compliance with the alternative shutdown option for those licensees that had not provided the information requested for staff review by the extended date granted to the licensee.

Second, requests for time extensions and exemptions from technical requirements were beginning to arrive at a fairly substantial rate. Forty-two plants had requested an extension of time (after March 19, 1981) to submit required information to NRC. The staff had intended to allow an extra year for 32 of these plants. However, the remaining ten had asked for dates ranging from March 30 to October 1982, and no schedule extensions had in fact been approved as of the end of September 1981. Also, as of October 1, 1981, 61 technical exemptions had been requested, of which 31 were still under review, nine had been denied, and 22 had been granted or found not to be required. The staff also intended to propose to the Commission a means to expedite data submitted under penalty of enforcement action.

In August of 1982, the Commission asked the staff to summarize the licensees' fire protection exemption requests, the staff's disposition of those requests, and any generic issues that had been raised by these requests. The Commission

also requested a description of the types of exemptions requested, the safety significance of the requests, a summary of the research results obtained, including a discussion of the impact that those results might have on fire protection requirements, and the need for revisions, if any, to the present fire protection requirements.

The staff response indicates that some problems were experienced with proper implementation of Appendix R because of differences between the staff and the licensees in interpretation of these requirements. As a result, the staff developed positions to clarify several of these requirements. These staff positions were sent to all licensees, construction permit holders and applicants for construction permits, for information purposes.

By May 31, 1983, staff review of operating plants for conformance to Appendix R was essentially complete. A total of 588 technical exemptions had been requested, of which 234 had been approved. Exemptions were approved on six of the 15 sections of Appendix R. As of mid-1983, the staff believed that most, if not all, backfits had been made to operating plants.

Fire Protection in New Plants

During the rulemaking proceeding, the Commission requested that the staff develop a fire protection rule for new plants by June 1982. This date was eventually extended to June 1984, to allow time for the staff to consider the impact of the fire protection program evaluations, including the exemption requests from Appendix R, as well as impact of the results of the ongoing research on the fire protection guidelines.

As of mid-1983, the staff believes that it may not be necessary to develop a comprehensive fire protection rule for new plants, for several reasons.

1. The fire protection requirements for new nuclear plants are set out in the Standard Review Plan, Section 9.5.1, "Fire Protection," Rev. 3, July 1981, which incorporates all the technical requirements of Appendix R.

2. Research results obtained to date have not demonstrated any inadequacies in the current fire protection requirements or guidelines.*

3. For new applications, including recent standard plant designs, redundant systems needed for safe shutdown must be separated by a three-hour barrier in all plant areas except in the control room and inside containment to eliminate complete reliance on active fire suppression systems. Capability to achieve safe shutdown independent of the control room is also required.

4. The exemption evaluations, research results, case work reviews and meetings with industry representatives have identified no new generic issues. The major effect on current requirements is the need for further clarification (provided by staff positions) and to remove requirements that have been found to be unnecessary. The staff intends to revise SRP 9.5.1 as soon as time and resources permit, with a target of September 1984, when all Appendix R exemption reviews should be completed.

5. Licenses being issued to new operating plants contain conditions through which the inspectors can verify that the licensee is meeting all the fire protection commitments.

However, the issue of a rule for new plants is not closed. The staff plans to review the necessity of developing a rule for future plants, and make recommendations by January 1, 1984. There still appears to be some disagreement within the staff as to whether a rule for new plants (licensed for operation after January 1, 1979) is necessary, though, as indicated above, the majority seemed comfortable relying on the BTP and standard review plan. Those who continue to favor a rule reason that a rule is stronger than a staff position and leads to more uniform enforcement. Further, the period of time between issuance of an operating license and actual operation is often substantial and sometimes involves continued construction or modification within the plant.

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*Available documentation did not indicate whether Appendix R took the results of the Sandia tests into account.

This "buffer period" tends to reduce the degree of regulatory influence over fire protection features.

3.2 Anticipated Transients Without Scram (ATWS)

Summary

This issue appears to be on the verge of resolution after more than 13 years. Its history reveals considerable disagreement between NRC and the reactor vendors, including unwillingness to provide data and in some cases overt pressure applied on the Commission by vendors. Lack of clear, justified guidance on the part of NRC contributed to vendor's unwillingness to make hardware changes. The question of whether to use rulemaking or compliance orders again arises. It is also clear that the lack of good analysis tools made the problem much more difficult to solve, an issue that continues to be of concern in handling common mode failures. It is also interesting that the use of a numerical safety goal was considered and finally rejected as inappropriate, at least in the context of ATWS.

Background

Anticipated transients without scram can be considered as a two-part occurrence. The first part, anticipated transients, concerns off-normal events that may occur during operation, including loss of electrical load, loss of feedwater, and loss of coolant flow. Some transients are fairly trivial, but others can impose substantial demands on plant equipment.

The reactor protection system is designed to detect off-normal situations and automatically shutdown - "scram" - the reactor. If a transient occurs and the reactor does not scram, an ATWS event has occurred. Some ATWS events could result in melting of the fuel and the release of large amounts of fission products. The question is whether the probability of such events is great enough to justify their consideration and, if so, what degree of protection is required.

Early History

ATWS has been the subject of extensive and continuing controversy among members of the nuclear industry and the regulatory staff, stemming mainly from differing perceptions of the potential extent and probability of serious consequences resulting from such events. Discussions of ATWS began as early as 1969, when a consultant to the ACRS pointed out the possibility of a safety problem if common mode failures could reduce the reliability of protection systems in such a way that the system might not function properly in the event of an anticipated transient. Even before this, the ACRS had been considering the degree to which systems to control the reactor and those to shut it down should be separated. Westinghouse, in particular, resisted complete separation, and stated that if forced to take this route, it would go to a single channel for control, which could lead to less reliable control and more transients. As a result of these concerns and ensuing discussions among the staff and ACRS, in late 1970, the staff requested the reactor manufacturers to prepare studies of the probability and consequences of a failure of the control rods to insert following an anticipated transient. Ten transients were analyzed for PWRs and six for BWRs. These studies concluded that the probability of such an event was negligibly low, but could result in failure of the reactor vessel or piping due to overpressure. Based on the review of these studies and other material, in September 1973 the staff published the "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" (WASH-1270) containing its position on ATWS. The ACRS had agreed with the position in April 1973..

The thrust of the staff position in WASH-1270 was that, because techniques for analyzing common mode failures were not as well developed as those dealing with random failures, the needed reliability of reactor shutdown systems was difficult to verify and, since larger safety margins were appropriate as increasing numbers of reactors were built and operated, ATWS events should be considered in the safety analysis of nuclear reactors. A safety objective of less than one chance in one million per year of a serious accident occurring at any individual plant was established as an "aiming point" based on an anticipated 1000 nuclear plants in the U.S. by the end of the century. For the particular failure path of ATWS, the staff set an objective of

one-tenth the overall safety goal. For future plants, for which applications for construction permits would be docketed after October 1, 1976, the staff concluded that an additional separate and diverse reactivity shutdown system should be required. For plants with construction permit applications docketed after early 1968, but before October 1, 1976, the staff concluded that any necessary design changes should be made to assure that the consequences of anticipated transients would be acceptable in the event of a postulated failure to scram. Later, the staff and the ACRS concluded that the requirement for an additional shutdown system in future plants was not necessary in order to achieve the desired safety objective and that the requirement to mitigate the consequences of ATWS events that applied to current plants would also be sufficient for new plants.

After WASH-1270 was issued, reactor manufacturers, in conjunction with the staff, began to develop methods to analyze ATWS events. A draft ANSI standard was written, outlining general guidelines for the analysis of ATWS events in PWRs. In October 1974 the vendors submitted reports describing the analysis of ATWS events for their reactor designs. At least some of these reports were not considered acceptable because they did not meet the intent of WASH-1270 or the guidelines, even though they showed system unreliability values of less than 10^{-7}. After requesting and receiving more information, the staff then found the vendor analysis methods to be generally acceptable, except for the treatment of system failures and some system parameters.

In mid-1976, applicants were requested to perform analyses for their plants using the methods developed by the vendors and modified as indicated in the staff status reports. These requests and the status reports led to substantial criticisms from applicants, reactor vendors and industry groups, stating that the staff requirements were unnecessary or at best overly conservative. The basic industry position was that the high reliability of reactor protection systems made the probability of ATWS event negligibly small and not worthy of consideration as a design basis. Industry also maintained that if ATWS events had to be considered in reactor safety evaluations, the requirements expressed in the staff status reports were excessively conservative. Such views were expressed in letters from individual applicants and industry groups, including the Atomic Industrial

Forum and the Electric Power Research Institute. Three of the vendors submitted additional reports in support of such arguments and provided evaluations of the reliability of their reactor protection systems, particularly with respect to the potential for common mode failures in the control rod drive mechanisms. EPRI also submitted a set of reports containing a detailed statistical analysis of scram system failure and the frequency of anticipated transients.

In early 1976, the ACRS endorsed the general approach and safety objectives adopted by the Staff, including the use of a goal of 10^{-7} per reactor year as the maximum probability from all causes of an ATWS with unacceptable consequences. However, the ACRS indicated there was a need for resolution of the arguments outstanding between vendors and staff concerning what constituted an acceptable way of meeting the criterion.

So, ATWS seemed to be "almost resolved." However, there remained many complicating factors. Many representatives of the nuclear utilities and the reactor vendors turned to the results of draft WASH-1400, issued in August 1974 and the final version in 1975, as a strong demonstration that ATWS was not a major contributor to risk from LWRs, and that hence the existing situation was satisfactory with no modifications, either in reliability of the scram system or in the ability of the reactors to accept an ATWS without intolerable consequences.

General Electric concluded that implementation of the mitigating requirements defined in the staff status report of December 9, 1975 would be very expensive,* and in a document dated September 30, 1976, proposed a different recommended solution to ATWS, namely incorporating a proposed alternative reactor scram system as backup to the existing system so that NRC's ATWS safety objective of 10^{-7}/reactor year could be met.

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*In a letter to the NRC dated September 29, 1976, the Long Island Lighting Company stated that "an expenditure of approximately $50 million would be required for the total installed cost of additional equipment and logic, but excludes cost of financing, outages or delays."

In a memorandum dated November 24, 1976, the ACRS consultant who orginally raised the ATWS issue discussed the General Electric proposal and continued to be unconvinced that the claimed reliability could be achieved. And, apparently the Regulatory Staff also remained unconvinced.

ATWS remained a very controversial issue between the NRC and the industry. In the fall of 1976, a series of reports entitled "ATWS: A Reappraisal" was published by the Electrical Power Research Institute (EPRI). The EPRI report re-evaluated the probability of failure to scram and estimated the risk to the public from ATWS. With their assumptions and data, the authors of the reports concluded that the probability of failure to scram was orders of magnitude lower than 10^{-4}/demand, and that ATWS posed an insignificant risk to the health and safety of the public.

The Regulatory Staff meet with EPRI in February of 1977 to discuss the EPRI reports. The Staff concluded that the probability of scram system failure due to a common mode failure was approximately 10^{-4} per reactor year based on experience, including some Navy data and reasserted the staff judgment that protection from ATWS events was required. The staff further noted that the probability of unacceptable consequences from ATWS events was also approximately 10^{-4} per reactor year. The Staff also had negative comments on the EPRI methodology. The EPRI representatives did not agree with all the staff comments.

However, consultants to the ACRS also supplied comments critical of specific aspects of the statistical treatment in the EPRI reports and the assumptions concerning treatment of data. In effect, they argued that the EPRI study had devoted considerable effort to a re-evaluation of scram unreliability as it applied in the case of a LOCA, not ATWS, and that the study was not relevant to ATWS.

In March 1977 the NRC established the first of two task forces on ATWS in an effort to finally resolve the matter. Meetings were held with reactor vendors. And in July 1977 the staff reported once again to the ACRS, reiterating their general position of December 1975 that scram unreliability could not be shown to be acceptably low and that measures were required to mitigate the consequences of ATWS.

It is somewhat curious, given this relatively unchanged position by the Staff over a period of many years, that little was done to implement any actual changes with regard to ATWS. For BWRs, for which the question first arose and for which at least one mitigating design change (the recirculating pump trip) was included as a partial backup to scram in all BWRs receiving construction permits after 1971, many if not most of the operating BWRs had not incorporated this feature by 1976. On learning of this situation in early 1976, the ACRS advised that steps be taken immediately to include this change in operating BWRs, unless it was not needed to make the consequences of ATWS tolerable. And the regulatory staff shortly thereafter issued letters to all operators of BWRs asking for their specific plans to incorporate this feature. However, by mid-1978 most of the operating BWRs had not implemented pump trip.

In April 1978 the regulatory staff issued a new report, NUREG-0460, entitled "Anticipated Transients Without Scram for Light Water Reactors." NUREG-0460 concluded that:

> "Considering the expected frequency of occurrence of transients, the reliability of current reactor scram systems necessary to meet the safety objectives has not been demonstrated and may well have not been attained. Therefore, we recommend that means of reducing the probability or consequences of ATWS events should be provided. Furthermore, we envision that the initiation of rulemaking to incorporate ATWS requirements in the Commission's regulations would fairly and clearly resolve the long standing uncertainty in the status of regulatory requirements in this area."

From early reactions to the staff position in April 1978, it appeared difficult to ascertain whether ATWS was finally headed toward early resolution. The reactor vendors, the utilities and the Electric Power Research Institute vigorously opposed the staff proposals.

In late 1978, after issuance of the report of the Risk Assessment Review Group concerning WASH-1400, the regulatory staff issued a revised position on ATWS, in Volume 3 of NUREG-0460, one which strongly reflected the difficulties in

backfitting an operating plant or even a plant under construction. That is, for such plants, emphasis was placed on those changes in circuitry that might provide increased scram reliability, while for plants still to be built, the emphasis was shifted to hardware changes to mitigate the consequences of an ATWS should it occur; that is, to keep pressures and temperatures below acceptable limits. The staff also moved away from the use of numerical safety objective, stating that, "We now believe that the resolution of ATWS concern should rest on engineering evaluation and judgment of the appropriateness of alternative plant modifications, rather than rest on quantitative risk analyses, though these should supplement staff judgment." Adequate verification of scram system reliability was still an issue and the acceptability of the hardware alternatives was to be subject to verification of the assumed level of protection offered by the modifications.

During 1979, generic questions and information guidelines were provided to the industry in a February letter. Originally, the answers to these 58 pages of questions and requests for information were scheduled for April 1979. A series of discussions resulted in separating the information needs for plants receiving a construction permit before January 1, 1978 into three categories, and rescheduling the answers for May-October 1979. Information about all other plants was deferred.

The Three Mile Island accident in March 1979 forced deferral of all NRC work on ATWS, and most industry work was halted or delayed as well. As a result, the information submitted by the industry fell far short of the February 15, 1979 request, and the staff concluded that the adequacy of the proposed modifications could not be demonstrated. In addition, industry still did not believe ATWS merited much concern, and if any action was to be taken, preferred only plant-specific modifications aimed at reducing the likelihood of ATWS events. Volume 4 of NUREG-0460, issued in March of 1980, included a list of areas inadequately addressed in the industry submittals.

This volume also set out a list of requirements proposed for licensees and applicants. But finally, movement toward a rule was becoming clear. In Volume 3, the staff had proposed rulemaking to establish general requirements

for all classes of plants; guidance on implementation of general criteria in a Regulatory Guide to be issued after an effective rule was in place; and early, generic verificiation of the adequacy of hardware modifications to achieve the levels of safety required by several alternative approaches. Now, in Volume 4, in view of the staff's evaluation that early, generic verification had not been achieved, the staff concluded that the alternatives that mandated prompt improvements in safety would best be accomplished by Orders. For a final and more controversial alternative that required additional verification and had a longer time span, the staff recommended rulemaking.

Even with this progress, a rule was not proposed until November of 1981.

The Proposed Rules

During the last few years of the development of a Commission position on ATWS, the industry had formed an owners group of 22 utilities to consider its position on this controversial subject. On September 16, 1980, the Group filed a petition for rulemaking, proposing a rule that would reduce the risk of ATWS by a factor of at least two. The proposed rule also recommended that the residual risk of ATWS be further considered in the ongoing, post-TMI degraded core rulemaking. The Group's reasoning was based on repeated statements by the NRC Staff that the current risk of ATWS was acceptable. Since the utilities' proposed rule would cut that risk at least in half, and since the number of power reactors could not conceivably double in the near term,* the utilities' rule would result in an overall lowering of risk below even what the NRC Staff considered acceptable. Partly in response to an incident at Brown's Ferry Unit 3 in June 1980, in which part of the control rods failed to fully insert, a supplement to the petition was filed on January 5, 1981, with detailed criteria for evaluating the scram discharge volume systems in boiling water reactors (BWRs). The petition and the supplement were posted in the Federal Register on November 4, 1980, and February 3, 1981, respectively.

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*A sharp contrast with the earlier-proposed NRC risk guideline based on approximately 1000 operating reactors by the year 2000.

During the ten months since the petition was filed, the Commission continued to grapple with the ATWS issue. The Office of Nuclear Reactor Regulation staff produced a proposed rule in several versions and, as recently as June 1981, then-Chairman Hendrie and the Office of Nuclear Regulatory Research produced an entirely new proposed rule.

At the meeting of the Commission on June 16, 1981, it was decided to publish the Research Office rule and one version of the NRR rule together, as alternatives, in the Federal Register. This raised the question of what was to be done with the Utility Group on ATWS rule. It appears from the transcript of the June 16 meeting that the Commissioners intended the utilities' rule at least to be mentioned in the Federal Register notice. But for several persuasive reasons, ranging from logical consideration of all available alternatives being to everyone's advantage to the strong suggestion that the utilities' rule was more cost effective, the Group urged the Commission to include the utilities' proposed rule as a third alternative.

The Commission granted the Group's request, and published on November 25, 1981, three proposed alternatives for ATWS, the comment period to expire on April 23, 1982. Supplementary information published with the proposals offers an interesting summary of the staff view at that time:

> "ATWS accidents are a cause for concern because a mismatch can develop between the power generated in the reactor and the power dissipated in controlled ways if the scram system fails to shut down the reactor following a fault in the normal heat dissipation functions (transient events). The power mismatch can threaten the integrity of the barriers that confine the fission products. A core meltdown accident, in some cases accompanied by a failure of containment and a very large release of radioactivity, is a possible outcome of some ATWS accident sequences. Thus, the consequences of some postulated ATWS accidents are unacceptable.
>
> There have been roughly one thousand reactor years of experience accumulated in foreign and domestic commercial light-water-cooled reactors without an ATWS accident. This experience suggests that the frequency of ATWS accidents is less than or of the order of once in a

thousand reactor years. There have been several precursor events, i.e., faults detected that could have given rise to ATWS events. This suggests that the frequency of ATWS accidents, though less than once in a thousand reactor years, may not be very much less. Such frequencies are too high for accidents of the severity described above. Thus the NRC has determined that reductions must be made in the frequency, severity, or both the frequency and severity of ATWS accidents."

The three proposals offer interesting comparisons of style and history. While available documentation is not totally comparable, the three proposals are briefly described below.

The first NRC proposed rule (the "staff rule") stems directly from NUREG-0460, Volumes 1-4, submitted in final form for the Commission's consideration on November 7, 1980. A basic consideration was that the two to four years needed for implementation offered an acceptably small risk of an ATWS event, even though at least one common cause failure in the reactor trip system was stated to have occurred (in about 1000 reactor years of experience) and several other common cause failures had occurred in "other systems." The conclusion was that the reliability of reactor protection systems had not been demonstrated and "most likely is not adequate." Thus, the probability of consequences of ATWS events would have to be reduced.

The Staff Rule (46 FR 57521) would have resolved ATWS by establishing performance criteria (each plant would have to perform analyses to verify that Service Level C of the ASME Boiler and Pressure Vessel Code would not be exceeded, fuel integrity would be maintained, there would be no excessive radioactivity release, the containment would not fail, and long-term shutdown and cooling would be assured). Three alternatives for increasing degrees of improvement in the reactor protection system and the capability to mitigate ATWS event consequences were included, basically geared to the vintage of the plant and to the type of vendor. Further discussion indicated that the staff recognized that the conclusion that an increase in the capability to mitigate ATWS events would provide a significant increase in the level of safety was still neither totally provable nor certain.

The Hendrie rule (46 FR 57521), while using much of the same information base as the staff rule, proposed to resolve ATWS by establishing a reliability assurance program for systems that prevent or mitigate ATWS accidents and prescribing certain hardware modifications which would allow for: (1) automatically tripping the recirculation pumps of a BWR under conditions indicative of an ATWS; (2) automatically actuating the standby liquid control system (SLCS) for BWRs; (3) providing a reliable scram discharge volume for BWRs; (4) providing for the prompt, automatic initiation of the auxiliary feedwater system for conditions indicative of an ATWS; and (5) assuring that the instruments necessary for the diagnosis of and recovery from ATWS accident sequences would not be disabled.

Finally, the utility rule proposed specific design modifications for each reactor manufacturer. It contained proposals that: (a) all Westinghouse reactors have initiation of the auxiliary feedwater system and turbine trip diverse from the reactor protection system; (b) all Combustion Engineering and Babcock and Wilcox reactors have diverse initiation of auxiliary feedwater and turbine trip (similar to Westinghouse) and a diverse scram system; and (c) existing boiling water reactors manufactured by General Electric have (1) a means to trip the recirculation pumps upon receipt of a signal indicative of an ATWS, (2) a diverse scram system, and (3) a modification of the scram discharge volume. Also, new (three years after the rule became effective) General Electric plants would have a standby liquid control system with increased flow and all reactor licensees would institute training for operators.

Forty public comments were received. Only one (a private individual) favored the Staff rule, though two utilities preferred "alternative 2A" of NUREG-0460, which was very similar to the utility rule. Most felt the Staff rule was too costly and time-consuming. Eleven respondents preferred the utility rule, six favored the Hendrie approach (though with many reservations, since this rule was considered "hard to interpret"), and nineteen favored no rule at all.

The comments provided by the utility group on ATWS consisted of a three-volume technical report which included a review and evaluation of past NRC and industry studies, a generic but substantial probabilistic risk assessment of

the issue for each reactor vendor, and a value-impact analysis of all three proposed rules. The group concluded that:

* The staff and Hendrie rules failed the value-impact test.

* Only the utility rule was consistent with current NRC policies.

* The record and notice for the staff and Hendrie rules were inadequate.

Through Sandia Laboratories, the Staff evaluated the utility group report and then, in July of 1982, formed a task force, composed of NRC personnel from several offices, to consider four alternatives:

1. Promulgation of no ATWS rule or including ATWS under the severe accident program;

2. Adoption of the proposed or a modified version of the utility group rule;

3. Adoption of the staff rule or a modification of it; or

4. Adoption of those portions of the Hendrie rule for which there exists a technical basis.

The "Final" Rule

On July 19, 1983, Commission approval was requested for publication in the Federal Register of a final ATWS rule and a related proposed rule. Over the previous year, the Task Force had reviewed all the comments and the utility group report, using a combination of probabilistic risk assessment (PRA) techniques and engineering judgment. The PRA techniques were used to develop and compare cost/benefit ratios for candidate regulatory changes and in developing a set of changes intended to ensure an acceptably low risk from ATWS events. The results of the PRA supported the staff's engineering judgment that rules were needed to reduce the risk from ATWS events.

At this point, the staff recommended publication of a final rule and related proposed rule. The final rule adopted the same approach that was used in the utility group's petition for rulemaking in that the prescribed changes are keyed to the reactor's type and manufacturer. With three exceptions, the prescribed changes in the final rule were proposed in the Utility Group's petition. Those exceptions were the new requirements for the standby liquid control system (SLCS) used on boiling water reactors and the inclusion of reactor trip breakers for the diverse scram system on pressurized water reactors manufactured by Combustion Engineering and Babcock and Wilcox.

The final rule also required an increased capacity for the SLCS on all BWRs. (The group had proposed the increased capacity for SLCS only for BWRs that are yet to be constructed.) The final rule also required that SLCS initiation be automatic (in lieu of manual initiation by the operator) for new plants and for other plants that have already been designed to include this feature. Automatic initiation of SLCS was not proposed by the group. The staff felt the SLCS was sufficiently important to reactor safety to require the increased capacity for both current and future reactors, and that it would be cost effective to require automatic initiation of SLCS for future reactors. The staff also believed that the inclusion of a diverse method of interrupting power to the diverse scram system was warranted based on the potential of common cause failure.

The proposed rule stemmed directly from the recent ATWS events at the Salem nuclear power plant. Salem uses PWRs manufactured by Westinghouse. The Salem events caused reconsideration of the need for diversity in scram systems in Westinghouse PWRs. The staff felt that the final rule's diversity requirement for Combustion Engineering and Babcock and Wilcox PWRs should also apply to Westinghouse PWRs and should also include the reactor trip breakers. However, the proposed rule also requested public comment on an alternative that would require a diverse scram system for Westinghouse reactors, but which would not require diverse turbine trip and auxiliary feedwater initiation.

The staff explicitly stated that the advantages of promulgating rules, rather than requiring changes on a case-by-case basis, were that the need for individual analyses would be eliminated and that "the two rules could lead to

completion of the rulemaking proceeding and to maintaining an acceptable level of risk from ATWS events without the expense to the NRC and the industry of plant-specific evaluations." The major disadvantages were that variations among plants might result in varying effectiveness of the rule and that possibly better ways of achieving safety would be discouraged.

Resource considerations seem to have been a major factor. The staff stated that implementation of the rules would have only a minor impact on NRC resources. ATWS considerations were already being reviewed by the staff on a case-by-case basis, and ATWS had become a contested issue by intervenor groups. Promulgating rules should reduce the amount of staff time currently required to communicate with applicants and intervenors on the issue.

Initially, little effort would be required to implement the rules. The Office of Nuclear Reactor Regulation would need to agree with vendors on acceptable design modifications, and implementation schedules had to be worked out with licensees. The regional offices were to be responsible for assuring that the modifications were properly installed.

Implementating the rules was estimated to cost licensees an estimated $3.5 million to $5.5 million per plant. The estimated total cost to industry is on the order of $525 million.

It appears that ATWS may be on the verge of at least some form of resolution, though the ACRS, in its July 1983 meeting, made a number of recommendations dealing only with the final rule (it will comment later on the proposed rule covering the Westinghouse trip system). These recommendations dealt primarily with environmental qualification of shutdown instrumentation, reducing the number of challenges to the reactor trip system, and the possible need to make some of the hardware changes required by the rule for safety grade equipment.

APPENDIX

Questions Used in Telephone Interviews

A. Identification of Safety Concerns

1. What avenues are available to raise safety concerns?

   a. NRC staff

   b. ACRS

   c. Nuclear critics

   d. Utilities

   e. Industry

2. What features of the current system inhibit the raising of safety concerns; i.e., timing, rules of evidence, etc.?

3. What safety improvements can be traced directly or indirectly to:

   a. The hearing process

   b. NRC staff

   c. ACRS

   d. Nuclear critics

   e. Utilities

   f. Industry

B. Resolution of Safety Concerns

1. How do issues once raised either get resolved or added to the list of unresolved safety issues?

   a. Is there difficulty with access to the decision-making process, including the establishment of schedules?

   b. What is the proper role of engineering judgement versus probabilistic risk assessment in resolving issues like backfitting?

2. Does the current method for resolving generic versus plant specific safety concerns lead to expansion of requirements as the resolution is being developed?

3. Are questions concerning the interactions of backfits with other plant systems adequately addressed in reolving safety concerns?

4. How much does resolution of issues depend on the competence, independence, and perspective of the NRC staff?

C. Safety Improvements

1. Have the backfits and other changes in regulatory requirements resulted in improvements or detriments to safety? Give examples of each.

2. Give examples of ways in which the regulatory process is arbitrary and capricious.

D. How to Change the System

1. What is a rational and convincing way to determine whether or not a specific backfit improves safety?

2. What would you propose changing to make the regulatory process work better to improve safety?

   a. Would you advocate more or less self-regulation by the utilities?

   b. How should organizations like INPO be used?

   c. If you were to design a new regulatory process, what form would it take?

3. How should the NRC deal with utilities who seem to be unable or unwilling to improve safety performance?